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Design of the Reactor Core for Nuclear Power Plants

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DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

FOREWORD

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CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide was prepared in support of safety requirements established in the Specific Safety Requirements publication, IAEA Safety Standards Series No. SSR-2/1, Safety of Nuclear Power Plants: Design [1], which was published in 2012 and was revised (as Rev. 1) in 2016. The present publication supersedes the Safety Guide on Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.12, published in 2005.

OBJECTIVE

1.2. The objective of this Safety Guide is to provide recommendations for safety in the design of the reactor core for nuclear power plants. This Safety Guide provides recommendations and guidance for the implementation and interpretation of safety requirements stated in the Specific Safety Requirements publication SSR-2/1 Rev. 1 [1], applied to the reactor core design.

SCOPE

1.3. This Safety Guide is intended for application primarily to land-based stationary nuclear power plants with water cooled reactors for electricity generation or for other heat production (such as district heating or desalination). All statements are applicable to light water reactors, i.e., pressurized water reactors and boiling water reactors, and are generally applicable to pressurized heavy water reactors unless otherwise specified. This Safety Guide may also be applied, with judgement, to other reactor types (e.g., gas-cooled reactors, floating reactors, small and ~~medium~~ modular ~~sized~~ reactors, innovative reactors) to provide interpretation of the requirements that have to be considered in developing the design.

1.4. The reactor core is the central part of a nuclear reactor where nuclear fission occurs. The reactor core consists of four basic systems and components, ~~(~~i.e., fuel (including fuel rods and fuel assembly structure), coolant, moderator and control rods~~)~~, and additional structures (e.g., reactor pressure vessel internals, core support plates, lower and upper internal structure in light water reactors). This Safety Guide addresses the safety aspects of the core design and includes the neutronic; thermalhydraulic; thermal mechanical; structural mechanical; reactor core control, shutdown and monitoring; and core management aspects for the safe design of the reactor core for nuclear power plants. Specifically, the following structures, systems and components are covered (see Annex II, Fuel, for terminology clarification):

(a) Fuel ~~elements~~ rods that include fuel pellets with or without burnable absorbers in cladding tubes, and generate and transfer heat to the coolant;

(b) Fuel assemblies that include a bundle of fuel ~~elements~~rods, along with structures and components (e.g., guide tubes, spacer grids, bottom and top nozzles, fuel channel) that maintain the fuel ~~elements~~ rods and fuel assemblies in a predetermined geometrical configuration;

(c) Reactor core control, shutdown and monitoring systems, including components and equipment used for reactivity control and shutdown, comprising the neutron absorbers (solid or liquid), the associated structure and drive mechanism;

(d) Support structures that provide the foundation for the core within the reactor vessel (within the Calandria in pressurized heavy water reactors), the structure for guiding the flow and guide tubes for reactivity control devices (for pressurized water reactors);

(e) Coolant;

(f) Moderator; and

(g) Other core components such as steam separators (in boiling water reactors) and neutron sources. These are considered only to a limited extent in this Safety Guide.

1.5. This Safety Guide is intended mainly for application to natural and enriched UO_2 fuels, and plutonium-blended UO_2 fuel (mixed-oxide fuel) with zirconium-based alloy cladding. Unless otherwise specified, all statements apply to these fuel types.

1.6. For innovative fuel materials (see Annex II, Fuel, for terminology clarification), such as uranium-nitride fuel or inert matrix fuel, or cladding materials other than zirconium-based alloys, this Safety Guide could be applicable with judgment.

1.7. The design of the reactor core may be interfaced with other reactor systems and other related aspects. In this Safety Guide, the guidance on these interfacing systems and aspects are described mainly to identify their functional considerations. Corresponding Safety Guides are referenced, as appropriate, in order to clarify their interfaces.

STRUCTURE

1.8. Section 2 describes general considerations for the safe core design based on requirements for management of safety, principal technical requirements and general design requirements established in Sections 3, 4 and 5 of ~~Ref.~~ [1], respectively.

Section 3 describes specific considerations for the safe design of fuel ~~elements~~rods, fuel assemblies, core structures and core components, and core control and reactor shutdown systems based on specific design requirements (i.e., Requirements 43 through 46) of ~~Ref.~~ [1].

Section 4 describes guidance on qualification and testing for the ~~core~~ structures, systems and components of the reactor core.

Annex I describes important items that need to be addressed within the design of the fuel ~~element~~rod, fuel assembly, reactivity control assembly, neutron source assembly and hydraulic plug assembly.

Annex II describes supplementary technical information for terminology clarification, additional background, or supporting examples for specified design recommendations.

2. GENERAL SAFETY CONSIDERATIONS IN THE DESIGN OF REACTOR CORE

MANAGEMENT SYSTEM

2.1. The design of the reactor core should be conducted taking into account the recommendations of GS-G-3.1 [3] and GS-G-3.5 [4] to meet Requirements 1-3 of SSR-2/1 Rev. 1 [1] and GSR Part 2 requirements [2].

DESIGN OBJECTIVES

Fundamental safety functions

2.2. The three fundamental safety functions, described in Requirement 4 of ~~Ref.~~ [1], are required to be ensured in the design of the reactor core, for operational states and a wide range of accident conditions. The fundamental safety functions are stated as follows for specific application to the design of the reactor core:

(a) Control of reactivity;

(b) Removal of heat from the reactor core; and

(c) Confinement of radioactive material.

Adequate design based on concept of defence in depth

2.3. Adequate design (i.e., capable, reliable and robust design) of the reactor core, based on the concept of defence in depth, will enable achievement of the fundamental safety functions, together with provision of associated reactor safety features.

2.4. Physical barriers considered as part of or affecting the design of the reactor core include the fuel matrix (see Annex II, Fuel, for terminology clarification), the fuel cladding and the boundary of the reactor coolant system ~~the reactor pressure vessel for light water reactors (or the primary heat transport system for pressurized heavy water reactors)~~. For normal operation and anticipated operational occurrences, fuel ~~elements~~ rods are required to be designed such that ~~to maintain~~ their structural integrity and a leaktight barrier (see Annex II, Cladding, for terminology clarification) are maintained to prevent fission product transport into the coolant (Requirement 43 of ~~Ref.~~ [1]). For design basis accidents and design extension conditions without significant fuel degradation, the reactor core is required to maintain a configuration such that it can be shut down and remain~~be~~ coolable (derived from Requirement 44 of ~~Ref.~~ [1]).

Proven engineering practices

2.5. The reactor core should be of a design that has been proven either by equivalent applications based on operational experience or on the results of relevant research programs, or appropriate according to the design and design verification/validation processes stated in applicable codes and standards (as indicated in Requirement 9, paras 4.14 and 4.16 of ~~Ref.~~ [1]).

Safety assessment in design process

2.6. As indicated in Requirement 10, para. 4.17 of ~~Ref.~~ [1], the safety assessment is required to be performed as part of the design process, with iteration between the design and confirmatory analyses, and increasing in its scope and level of detail as the design progresses. Guidance on safety assessment methods is described in ~~Ref.~~ [5].

Features to facilitate radioactive waste management

~~2.6.~~2.7. The design of fuel ~~elements~~ rods and fuel assemblies should account for features that will facilitate future waste management (including reprocessing when applicable). Physical conditions of discharged fuel assemblies from the reactor core affect the design of the storage and disposal systems for the used fuel. Guidance to account for the impact of used fuel conditions on the design of spent fuel handling and storage systems are described in ~~Refs~~ [6] and [7].

DESIGN BASIS FOR STRUCTURES, SYSTEMS AND COMPONENTS OF THE REACTOR CORE

~~2.7.~~2.8. As indicated in Requirement 14 of ~~Ref.~~ [1], the design basis for the reactor core are required to specify the necessary capability, reliability and functionality for all applicable plant states (see para. 2.9) in order to meet the specific acceptance criteria.

Categories of plant states and postulated initiating events

~~2.8.~~2.9. Plant states, as described in Requirement 13 of ~~Ref.~~ [1], which are typically considered for the design of the reactor core are normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation (these four states are called "all applicable plant states" throughout this Safety Guide). Design extension conditions with core melt are out of the scope of the reactor core design.

~~2.9.~~2.10. The design process should include analysis of the effects of postulated initiating events on the reactor core for all applicable plant states. Guidance on the identification of postulated initiating events for all applicable plant state and relevant analyses is described in ~~Ref.~~ [5].

External hazards

~~2.10.~~2.11. Consequences of earthquake should be taken into account in the design of the reactor core. Seismic categorization of the structures, systems and components of the reactor core should be determined according to ~~Ref.~~ [8].

Design limits

~~2.11.~~2.12. Design limits for individual structures, systems and components of the reactor core are required to be specified for all applicable plant states (as indicated in Requirement 15 of ~~Ref.~~ [1]). The fulfillment of these limits with appropriate provisions will assure that the concept of defence in depth stated in para.2.4~~2.4~~ is successfully implemented with adequate margins. Typical examples of these parameters with quantitative or qualitative limits are described in Section 3. (See Annex II, Margin, for terminology clarification.)

Safety classification aspects of the reactor core

~~2.12.~~2.13. The structures, systems and components of the reactor core are required to be classified on the basis of their function and their safety significance (see Requirement 22 of ~~Ref.~~ [1]). The safety classification process is described in ~~Ref.~~ [10].

~~2.13.~~2.14. F~~F~~uel ~~elements~~ rods and fuel assemblies should be identified as Safety Class 1 in ~~Ref.~~ [10], the highest safety class, since they are essential to achieve the three fundamental safety functions in para. 2.2. ~~. The leaktightness and structural integrity of fuel elements are required to maintain these barriers to the release of radioactive materials. Structural integrity of fuel assemblies is required to maintain geometry compatible with design basis, in particular, to ensure a coolable geometry during accidents.~~

~~2.14.~~2.15. Failure of control rods has the potential to endanger the leaktightness and structural integrity of the fuel ~~element~~ rod which is a Safety Class 1 barrier; from this perspective, control rods should be identified as a Safety Class 1 component.

~~2.15.~~2.16. For all Safety Classes identified according to the method described in ~~Ref.~~ [10], corresponding engineering design rules should be specified and applied.

Engineering design rules

~~2.16.~~2.17. The engineering design rules for the structures, systems and components of the reactor core represent methods to achieve the adequacy of the design and should include the following, as appropriate:

(a) Use of applicable codes and standards, and proven engineering practices;

(b) Conservative safety assessment;

(c) Specific design analyses for reliability;

(d) Qualification and testing; and

(e) Operational limits and conditions.

Design for reliability

~~2.17.~~2.18. According to Requirement 23, para. 5.37 of ~~Ref.~~ [1], fuel ~~elements~~ rods and assemblies, and components and systems for reactor control and shutdown are required to be designed with high reliability, in consideration of their safety significance. Provision for achieving high reliability in these designs is described in paras 3.27~~3.27~~ and 3.88~~3.88~~, respectively.

Operational limits and conditions

~~2.18.~~2.19. As indicated in Requirement 28 of ~~Ref.~~ [1], operational limits and conditions should be established in order to ensure that the reactor core operates safely, in accordance with design assumptions and intent. Relevant guidance on the operational limits and conditions is described in ~~Ref.~~ [11].

DESIGN FOR SAFE OPERATION

~~2.19.~~2.20. The structures, systems and components of the reactor core should be designed such that their required testing, inspection, repair, replacement, calibration or maintenance is facilitated.

~~2.20.~~2.21. The design of the reactor core should be reviewed and modified when a significant configuration change occurs during the plant’s operating lifetime, as a result of, for example:

(a) Major plant design, equipment or operational modifications, such as

- Replacement of steam generator (not for boiling water reactors),
- An increase in the rated power of the plant,

- A significant change in the operating domain, etc.;

(b) New fuel type or significant changes in fuel types (e.g., use of mixed-oxide or gadolinium fuel; new fuel rods or new fuel assembly designs with modified geometrical or thermal hydraulic characteristics~~A significant change in fuel types (e.g., mixed-oxide fuel~~);

(c) An increase of the discharge burnup beyond the design limit; and

(d) Major fuel management changes such as large cycle length extension. ~~A significant increase in the duration of a fuel cycle;~~

~~(d) An increase in the rated power of the plant; and~~

~~(e) A significant change in the operating domain.~~

~~2.21.~~2.22. Fuel ~~elements~~ rods and fuel assemblies should be designed to prevent the potential for fuel failures due to specific operational conditions (e.g., startup rates, degraded coolant chemistry conditions, or presence of foreign material) during operational states.

REACTOR CORE SAFETY ANALYSIS

~~2.22.~~2.23. As indicated in Requirement 42 of ~~Ref.~~ [1], safety analysis is required to evaluate and assess challenges to safety in all applicable plant states, using deterministic ~~-~~approaches including uncertainties to the extent possible.~~.~~

~~2.23.~~2.24. The following major factors should be accounted for in the reactor core safety analysis:

(a) Initial operating~~Operating~~ ~~state~~conditions (e.g., global and local thermalhydraulic conditions, power levels, power distributions and time in the cycle);

~~(a)~~(b) Reactivity feedbacks;

~~(b) (b) Temperature coefficient of reactivity for the fuel (Doppler coefficient);~~

~~(c) (c) Temperature coefficients of reactivity for the coolant and the moderator;~~

~~(d) (d) Void coefficients of reactivity for the coolant and the moderator;~~

~~(e)~~(c) Rate of change of the concentration of soluble absorber in the moderator and the coolant;

(d) Position or r~~R~~ate of insertion of positive (or negative) reactivity caused by the reactivity control device(s) or changes in process parameters;

(e) Rate of insertion of negative reactivity associated with a reactor trip;

(f) Individual channel transient response related to the average thermal power of the core (for boiling water reactors);

~~(f)~~(g) Performance characteristics of safety system equipment, including the changeover from one mode of operation to another (e.g., from the injection mode for emergency core cooling to the recirculation mode); ~~and~~

(h) ~~(j)~~ Decay of xenon and other neutron absorbers in the long term core analysis; and

~~(g)~~(i) ~~.~~Core activity inventory.

Appropriate provisions or margins should be included in the above factors such that the safety analyses remain valid for specific loading patterns or fuel designs. Guidance on safety analysis methods is provided in [5].

2.25. Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all applicable plant states. For accident conditions the effect of fuel behavior on core cooling should be included in the safety analysis, e.g., ballooning and rupture of the cladding, exothermic metal–water reactions and distortions of fuel ~~elements~~ rods and fuel assemblies. The effects of ~~Accumulation of~~ hydrogen accumulation ~~,~~ (as a result of a metal–water reaction between the zirconium-based alloy cladding and water at high temperature) on the boundary of reactor coolant system should be evaluated.~~, could threaten the integrity of the reactor pressure vessel for light water reactors (of pressure tubes for pressurized heavy water reactors) and of the containment and should be evaluated.~~

2.26. A systematic, complete, qualified and up-to-date documentation of the state of the structures, systems and components of the reactor core should be available in order to be able to perform the safety analysis on the actual plant configuration.

3. SPECIFIC SAFETY CONSIDERATIONS IN THE DESIGN OF THE REACTOR CORE

GENERAL

3.1. This section addresses specific design aspects for the structures, systems and components of the reactor core to meet Requirements 43–46 described in the Specific Safety Requirements publication SSR-2/1 [1]. It also includes the interface with core management which strongly influences the core design with regard to the performance of fuel ~~elements~~ rods and fuel assemblies. (Specific guidance for core management and fuel handling in nuclear power plants is described in ~~Ref.~~ [15].)

3.2. The design of the reactor core should enable the fulfillment at all times of the fundamental safety functions (para.~~2.1~~2.2) for all applicable plant states (i.e., normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation), in combination with reactor cooling systems, and reactor control and reactor protection systems.

3.3. The reactor core and associated control and protection systems should be designed with adequate margins to assure that fuel design limits are not exceeded during all applicable plant states. Fuel design limits are described in paras 3.49–3.59. ~~3.49-3.59.~~

Fuel type

3.4. Fuel ~~elements~~ rods contain fissile materials (e.g., U-235, Pu-239) that are highly reactive with thermal neutrons. Fuel pellet materials should be selected with consideration of the following optimized properties (examples of the pellet material are described in Annex II, Fuel):

(a) Reactivity with thermal neutrons;

(b) Impurities with low thermal neutron absorption properties;

(c) Thermal performance (e.g., high thermal conductivity for operational states and high thermal diffusivity for accident conditions are desirable);

(d) Dimensional stability;

(e) Fission gas retention; and

(f) Pellet-cladding interaction resistance.

3.5. Cladding materials should be selected with consideration of the following properties (examples of cladding materials are described in Annex II, Cladding):

(a) Low absorption cross-section for thermal neutrons;

~~(a)~~(b) High resistance to irradiation conditions;

~~(b)~~(c) High thermal conductivity and high melting point;

~~(c)~~(d) High corrosion resistance and low hydrogen pick-up;

~~(d)~~(e) Low oxidation/hydriding at high temperature conditions;

~~(e)~~(f) Adequate breakaway oxidation resistance at integrated-time temperature conditions. Integrated-time temperature refers to the assessment of total time achievable at a given cladding temperature without reaching oxidation breakaway (uncontrolled oxidation kinetics);

~~(f)~~(g) Adequate mechanical properties, e.g., high strength, ~~and~~ high ductility, low creep rate in normal operation and high relaxation rate in transients; and

~~(g)~~(h) Low susceptibility to stress corrosion cracking.

Coolant

3.6. In light water reactors, the coolant also acts as the moderator. The ~~design~~ choice of coolant should account for all potential interactions between chemical conditions in the coolant and fuel and core components (see Annex II, Coolant, for supplementary information).

For pressurized heavy water reactors, the coolant and the moderator are separated; typically, chemicals are not added to the coolant for controlling reactivity.

3.7. The coolant should be physically and chemically stable with respect both to high temperatures and to nuclear irradiation in order to fulfill its primary function: the continuous removal of heat from the core. (Fuel and core coolable geometry should be maintained and the reactor core should be designed to prevent or control flow instabilities and resultant fluctuations in core reactivity or power.) The reactor fuel and core design should also include the following safety considerations associated with the coolant:

(a) Ensuring that the coolant system is free of foreign materials prior to the initial start-up of the reactor and following refueling and maintenance outages for the operating lifetime of the plant;

(b) Maintaining the radionuclide activity in the coolant as low as reasonably achievable ~~at an acceptably low level~~ by means of purification systems, corrosion product minimization, or removal of defective fuel as appropriate;

(c) Monitoring and controlling the effects that the coolant and coolant additives have on reactivity under all plant states; ~~and~~

(d) Determining and controlling the physical and chemical properties of the coolant in the core; and~~.~~

(e) Ensuring that coolant chemical composition is compatible with materials which are present in the primary circuit (e.g., to avoid crud formation on fuel rods, to minimize corrosion and radioactive product generation, etc.).

3.8. The design should account for the effect of changes in coolant density (including fluid phase changes) on core reactivity and core power, both locally and globally.

Moderator

3.9. The choice of moderator and of the spacing of the fuel ~~elements~~ rods and fuel assemblies within it should meet engineering and safety requirements on the reactivity feedbacks due to changes in moderator temperature, density and void fraction, ~~coefficient of reactivity,~~ while aiming at optimizing the neutron economy and hence fuel consumption. The prevalent thermal reactor types use either light water or heavy water as the moderating medium.

Depending on the reactor design, the moderator could contain a soluble neutron absorber, such as boron in pressurized water reactors, to maintain adequate shutdown margins during operational states and to compensate the decrease in core reactivity throughout whole reloading cycle.

3.10. In pressurized heavy water reactors, the reactor core design should assure the effectiveness of the shutdown ~~and hold-down capability~~ system of the reactor during an absorber dilution accident. Means should be provided to prevent the inadvertent removal of such absorber material (e.g., due to chemistry transients) and to ensure that its removal is controlled and slow.

The moderator should provide the capability to remove decay heat without loss of core geometry for ~~all applicable plant states~~accident conditions (pressurized heavy water reactors).

Measures should be provided to prevent deflagration or explosion of hydrogen generated by radiolysis in the moderator (pressurized heavy water reactors).

NEUTRONIC DESIGN

Design considerations

3.11. The design of the reactor core should assure that the feedback characteristics of the core rapidly compensate for an increase in reactivity. The reactor power should be controlled by a combination of the inherent neutronic characteristics of the reactor core (see Annex II, Reactivity feedbacks, for supplementary information) and its thermalhydraulic characteristics, and the capability of the control and shutdown systems to actuate for all applicable plant states.

~~3.10.~~3.12. The design should assure that power changes that could result in conditions exceeding fuel design limits for normal operation and anticipated operational occurrences should be reliably and readily detected and suppressed.

Nuclear design limits

Nuclear key safety parameters

~~3.11.~~3.13. Nuclear key safety parameters influencing neutronic core design and fuel management strategies should be established from the safety analyses that verify the compliance with specific fuel design limits described in paras 3.49–3.59. ~~3.49-3.59.~~ Appropriate provisions should also be provided for the nuclear key safety parameters, such that they would remain valid for specific core reload designs and throughout whole reloading cycle. Typical nuclear key safety parameters include:

(a) Temperature coefficients of reactivity for fuel and moderator;

(b) Boron reactivity coefficient and concentration (pressurized water reactors);

(c) Shutdown margin;

(d) Maximum reactivity insertion rate;

~~s;~~

(e) Control rod and control bank worth;

(f) Radial and axial power peaking factors, including allowance for ~~Xe~~ xenon induced oscillation;

(g) Maximum linear heat generation rate; and

(h) Void coefficient of reactivity.

The safety impacts of any major modifications (see para.2.21) on the reactor core design should be assessed using the nuclear key safety parameters, in order to ensure that the specified fuel design limits are not violated. Otherwise, new nuclear key safety parameters should be defined and justified. ~~Such modifications could include:~~

~~(a) Major plant design, equipment or operational modifications;~~

~~(b) Major fuel management changes such as large cycle length extension;~~

~~(c) New fuel type introduction (e.g., mixed-oxide or gadolinium fuel); and~~

~~(d) Fuel burnup limit extension.~~

Core reactivity characteristics

~~3.12.~~3.14. On the basis of the geometry and the fuel composition of the reactor core, the design should include nuclear evaluations to provide steady state spatial distributions of neutron flux and of the power, core neutronic characteristics and the efficiency of the means of reactivity control for normal operation of the plant at power, shutdown and accident conditions.

~~3.13.~~3.15. Nuclear key safety parameters such as reactivity coefficients should be evaluated for selected core operating conditions (e.g., zero power, full power, beginning of cycle, end of cycle) and for the corresponding fuel management strategy. Their dependence on the core loading and the burnup of the fuel should be analyzed. Appropriate provisions should be included in the reactivity coefficients or other reactivity feedback modeling approaches used in the safety analysis for all applicable plant states.

Maximum reactivity worth and reactivity insertion rate

~~3.14.~~3.16. The maximum reactivity worth of the reactivity control devices (e.g., control rods and/or chemical and volume control system~~boron~~) should be limited, or interlock systems should be provided, so that any resultant power variations do not exceed specified limits for relevant reactivity insertion transients and accidents, such as:

(a) Control rod ejection;

(b) Control rod drop;

(c) Boron dilution; and

(d) Uncontrolled bank withdrawal.

These reactivity limits should be determined via safety analyses to ensure that fuel design limits described in paras ~~3.49–3.59~~ 3.49–3.59 are ~~met~~not exceeded. These analyses should be performed for all fuel types in the core (e.g., UO_2 or mixed-oxide fuel) or a representative core with appropriate provisions and as a function of allowable operating conditions and fuel exposure.

Control of global and local power

~~3.15.~~3.17. The design should assure that the core power can be controlled globally and locally using the means of reactivity control (see Annex II, Control, for example~~s~~) in such a way that the peak linear heat generation rate of each fuel ~~element~~ rod does not exceed the specified limits anywhere in the core. Variations in the power distribution (e.g., caused by effects like xenon instability) ~~local variations in reactivity due to xenon instability~~ or other local effects (e.g., mixed core, crud induced power shifts or axial offset anomalies for pressurized water reactors, fuel assembly bow or distortion) should be addressed in the design of the control system. Provisions should be included to account for measurement variations between flux detectors (e.g., due to operability, location, shadowing or ageing).~~.~~

Shutdown margin

~~3.16.~~3.18. The insertion of control rods should provide adequate shutdown margin (see Annex II, Margin, for clarification of ~~the~~ terminology) in all applicable plant states. The specification and monitoring of control rod insertion limits as a function of power level should assure adequate shutdown margin at all times.

~~3.17.~~3.19. The effects of depletion of burnable absorber~~-~~ on the core reactivity should be evaluated to ensure adequate shutdown margin in all resulting applicable core conditions throughout the fuel cycle. (Examples of using burnable absorbers in pressurized water reactors are provided in Annex II, Burnable absorber.)

THERMALHYDRAULIC DESIGN

Design considerations

~~3.18.~~3.20. The thermalhydraulic design of the reactor core should include adequate margins and provisions to assure that

(a) Specified thermalhydraulic design limits are not exceeded in operational states (i.e., during normal operation and anticipated operational occurrences);~~nces), and~~

(b) ~~(b)~~ The failure rates of fuel ~~elements~~ rods during design basis accidents and design extension conditions without significant fuel degradation remain below acceptance levels; and~~.~~

~~(b)~~(c) Minimal and maximal values of core flow rate are consistent with thermalhydraulic and mechanical design limits.

Thermalhydraulic design limits

~~3.19.~~3.21. Specific thermalhydraulic design limits should be established with adequate~~-~~ margins on predictable parameters, such as the maximum linear heat generation rate, the minimum critical power ratio (for boiling water reactors) or the minimum departure from nucleate boiling ratio (for pressurized water reactors) or dryout power ratio (for pressurized heavy water reactors), the peak fuel temperature or enthalpy, and the peak cladding temperature. Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculation methods used in the assessment of thermal margin should be addressed in the design analyses.

~~3.20.~~3.22. The thermalhydraulic design should include design analyses that take into account design features of a fuel assembly including fuel ~~element~~ rod spacing, fuel ~~element~~ rod power, sizes and shapes of subchannels, spacer and mixing grids (for light water reactors), and flow deflectors (for light water reactors) or turbulence promoters. In addition, for fuel channel type pressurized heavy water reactors, effects of fuel bundle string, appendages, gaps between fuel ~~elements~~ rods and pressure tube, anticipated shape changes of pressure tubes with reactor ageing, and junctions between neighboring endplates should be addressed in the design analyses.

For light water reactors, the thermalhydraulic design should also consider core inlet and outlet coolant temperature and flow distributions. These effects should also be considered in the core monitoring and protection systems.

The design should assure that the minimum ratio of operating power to critical power (i.e., a minimum critical heat flux ratio, a minimum departure from nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio) considers that critical heat flux correlations have been developed from representative tests performed at steady state conditions. As a consequence, adequate margins or provisions should be added to the minimum ratio to account for additional factors not considered in the correlation itself. Examples of these factors are:

(a) The thermal hydraulic response to anticipated operational occurrences;

(b) Impacts resulting from the chosen loading pattern;

(c) Impacts resulting from the potential presence of crud in the core.

In addition, uncertainties such as plant operational uncertainties and code uncertainties should be adequately taken into account in the safety analysis.

~~provided to allow for anticipated operational occurrences.~~ The objective is to avoid potential for cladding failure by using critical heat flux limits as surrogates.

In some reactor designs critical heat flux conditions during transients can be tolerated if it can be shown using suitable analytical methods that the cladding temperatures do not exceed the dryout induced fuel failure limits.

~~3.21.~~3.23. Experiments should be conducted on representative fuel assembly designs and over the range of expected operational ~~conditions~~ states, including various axial heat flux profiles, to identify ~~to provide data for defining~~ the limiting values of the minimum ratios. Correlations for predicting critical heat flux are continually being ~~revised~~ generated as a result of additional experimental data, changes in fuel assembly design, and improved calculation techniques involving coolant mixing and the effect of axial power distributions. ~~These correlations may be overly conservative for~~ For fast transients (e.g., rod ejection accidents), ~~and therefore, they~~these correlations may be reassessed as steady state conditions may be not sufficiently representative.~~for these applications.~~

~~3.22.~~3.24. Approaches such as those in the following examples should be taken to demonstrate the fulfillment of paras. 3.21–3.23.~~3.21-3.23:~~

(a) Regarding departure from nucleate boiling ratio, critical heat flux ratio or critical power ratio correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot ~~element~~ rod (see Annex II, Fuel, for terminology clarification) in the core does not experience ~~a departure from nucleate boiling or boiling transition condition~~ any heat transfer deterioration during normal operation or anticipated operational occurrences;

(b) For light water reactors, t~~T~~he limiting (minimum) value of departure from nucleate boiling ratio, critical heat flux ratio, or critical power ratio correlations should be established such that the number of fuel ~~elements~~ rods that experience a departure from nucleate boiling or boiling transition during normal operation or in anticipated operational occurrence conditions does not

~~exceed~~ exceed a specified limit, i.e., ~~e.g.,~~ at ~~least~~ most one ~~element~~ fuel rod per thousand in the ~~reactor~~ reactor core; ~~or~~

(c) For pressurized heavy water reactors, if the maximum fuel cladding temperature remains below a certain limit (e.g., 600 °C) and the duration of post-dryout operation is limited (e.g., less than 60 seconds), it is considered that the fuel deformation is small so that fuel elements are not in contact with the pressure tube and will not cause a failure of the pressure tube.

FUEL ~~ELEMENT~~ ROD AND FUEL ASSEMBLY THERMAL MECHANICAL DESIGN

Design considerations

3.25. The design should assure that the structural integrity of fuel assemblies (geometry) and fuel ~~elements~~ rods (leaktightness) is maintained for normal operation and anticipated operational occurrences. For accident conditions (design basis accidents and design extension conditions without significant fuel degradation) only a limited number of fuel failures should be allowed. The allowable number of failed fuel rods may depend on the frequency and nature of the event. ~~Long term~~ C~~c~~oolable geometry should be ~~maintained~~ensured by design for accident conditions. ~~-~~Under these conditions, the level of radionuclide activity should be assessed to confirm that the permissible dose limits ~~for the release of fission products~~ are met.~~.~~

In accident conditions that lead to cladding ballooning and rupture, fuel fragment dispersal in the coolant should be prevented.

~~3.23.~~3.26. The design of fuel ~~elements~~ rods (with or without burnable absorbers) and fuel assemblies should address the irradiation and environmental conditions (e.g., temperature, pressure, coolant chemistry, irradiation effects on fuel, cladding and fuel assembly structure; static and dynamic mechanical loads including flow induced vibration; and changes in the chemical characteristics of the constituent materials).

The design should also assure that fuel ~~elements~~ rods and fuel assemblies can withstand handling loads in transport, storage, installation and refueling operation.

Important items that are typically addressed in the design of fuel ~~elements~~rods and~~,~~ fuel assemblies from the irradiation and environmental aspects are described in Annex I, together with those for reactivity control assemblies, neutron source assemblies and hydraulic plug assemblies.

~~3.24.~~3.27. The design should assure that fuel ~~elements~~ rods and fuel assemblies ~~-~~are reliable throughout their lifetime including manufacturing, transportation, handling, in-core operation, ~~interim~~ storage, and disposal, where applicable. Several key contributors throughout their lifetime should be addressed; important key contributors to fuel reliability include:

(a) Fuel fabrication oversight;

(b) Debris mitigation (foreign materials exclusion);

(c) Control of in-reactor power changes to limit excessive pellet-cladding interaction;

(d) Control of crud and corrosion;

(e) Prevention of grid-to-~~fuel element~~ rod fretting (for light water reactors); and

(f) Fuel surveillance and inspection practices.

Thermal and burnup effects on fuel ~~elements~~rods

~~3.25.~~3.28. In operational states, the design should assure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. ~~In~~ For design basis accidents (e.g., reactivity initiated accidents) and for design extension conditions without significant fuel degradation, ~~limited~~ incipient fuel melting can be ~~allowable~~allowed- (e.g., fuel centerline melting is limited to a small fraction of fuel pellet volume).~~locally~~ ~~.~~ The design and safety assessments should account for burnup effects on the fuel rod and fuel assembly properties (see Annex II, Fuel, for supplementary information).

~~3.26.~~3.29. Straining of the cladding is caused by fuel ~~element~~ rod internal gas overpressure or by fuel gaseous swelling or fuel thermal expansion as a consequence of fuel burnup or local power increases. The design should assure that cladding stresses and strains are limited; limits for cladding stress, ~~long term deformation~~accumulated cladding strain -and cladding corrosion/hydriding should be specified for all applicable plant states and applied throughout whole reloading cycle. ~~different operational and accident states.~~

~~The consequences of significant~~ For accident conditions, cladding deformation ~~(e.g., cladding ballooning) during operational states~~ should be evaluated ~~in accident analyses~~ to determine the potential for cladding failure (e.g., burst or rupture) and any resulting release of fission products from the fuel.

Effects of irradiation on fuel assembly structures

~~3.27.~~3.30. The design should assure that the dimensional ~~stability~~ changes of light water reactor fuel assembly structures ~~is maintained~~are minimized~~,~~ so that contacts or interactions between ~~the~~ fuel rods and ~~the~~ fuel assembly components (top and bottom fuel assembly nozzles) are ~~avoided~~precluded, and that fuel rod bow and fuel assembly bow, as well as control rods swelling and any potential interaction with the fuel assembly guide tubes, do not affect the structural integrity or the thermalhydraulic performance of the fuel assemblies or ~~the performance of~~ the ~~control rod~~ safety functions of the control rod.

Grid spring relaxation under irradiation should be assessed to limit the potential for grid-to-~~fuel element~~ rod fretting. In the dimensional stability analyses, the effects of irradiation, in particular the effects of fast neutrons on fuel assembly components and control devices, on their mechanical properties such as tensile strength, ductility, growth or creep/relaxation should be taken into account. The effect of irradiation on buckling resistance of the spacer grids should be considered when assessing seismic events or loss-of-coolant accidents.

~~3.28.~~3.31. For pressurized heavy water reactors, the design should assure that the fuel cavity length in the fuel channel is sufficient to accommodate the irradiation and thermal effects on the fuel string in the fuel channel for all applicable plant states.

Effects of variations in power levels

~~3.29.~~3.32. For operational states, the design should assure that fuel ~~elements~~ rods withstand thermal mechanical loads during local and global power transients (e.g., due to fuel assembly shuffling, to movements of control devices, to load following, to flexible operation or to other causes of reactivity changes).

Mechanical effects in fuel ~~elements~~rods

~~3.30.~~3.33. The design should include analyses to assure that straining of the fuel cladding due to mechanical loads (e.g., coolant pressure, seismic loads) meet fuel design limits. The analyses should take into account the radial gap closure kinetics that depends on various parameters such as fuel densification~~, properties,~~ fuel ~~gaseous~~ swelling, fuel pellet cracking, fragmentation and its radial ~~repositioning~~relocation within the fuel rod after a power change~~ of fragments~~, cladding creep behavior~~s~~ at low stress~~es~~, initial ~~element~~ rod internal pressure, fission gas release to the free volumes, and operating parameters including power history and coolant pressure.

~~3.31.~~3.34. Stress corrosion cracking induced by pellet-cladding interaction in the presence of corrosive fission products should be minimized.

~~3.32.~~3.35. Stress concentration caused by missing pellets, axial gaps between fuel pellets, missing pellet surfaces or fuel pellet chips trapped in the gap cannot be explicitly considered in the fuel ~~element~~ rod design, and hence those anomalies should be avoided to the extent possible.

Effects of burnable absorber in the fuel

~~3.33.~~3.36. The design should include analyses to demonstrate that the fuel ~~element~~ rod can accommodate for the effects of any in-fuel burnable absorbers on fuel pellets' thermal, mechanical, chemical, and microstructural properties and on fuel ~~element~~ rod behavior.

Corrosion and hydriding

~~3.34.~~3.37. ~~The cladding design should ensure that~~ H~~h~~ydrogen pick-up correlation ~~is~~ should be ~~adequately~~ specified for each cladding type so that some fuel design limits, such as for reactivity initiated accident and loss of coolant accident, can be expressed as a function of the cladding pre-transient hydrogen content. (See Annex II, Cladding, for supplementary information.)

~~3.35.~~3.38. Fuel ~~elements~~ rods and fuel assemblies should be designed to be compatible with the coolant environment in operational states, including shutdown and refueling. (See Annex II, Cladding, for supplementary information.)

~~3.36.~~3.39. For pressurized heavy water reactors, initial hydrogen content in the fuel ~~element~~ rod should be limited to reduce the likelihood of fuel defects being caused by hydrogen induced embrittlement of the cladding.

Crud

~~3.37.~~3.40. The design analyses should account for the degradation ~~in heat transfer~~ of the fuel ~~element~~ rod heat transfer due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. In case boron is trapped in the crud layer in pressurized water reactors, its potential impact on the neutronic performance of the core should be assessed and addressed in the core design analyses.

Hydraulic effects in fuel assemblies

~~3.38.~~3.41. Hydraulic effects should be addressed primarily in the thermalhydraulic design of the fuel assembly, and in the evaluation of localized corrosion, erosion, flow-induced vibration, ~~and~~ grid-to-rod fretting, fuel assembly lift-off, ~~of the~~ fuel ~~assembly~~assembly distortion, etc. Hydraulic effects on the fuel assembly design should be ~~demonstrated~~ characterized via fuel assembly endurance tests performed in qualified out-of-reactor loops, using full scale fuel assembly mock-ups with prototypical test conditions (e.g., pressure, temperature, cross-flows and end-of-life grid spring relaxation).

Considerations of mechanical safety in the design

~~3.39.~~3.42. The fuel assembly should be designed to withstand mechanical stresses as a result of:

(a) Fuel handling and loading;

(b) Power variations;

~~(b)~~(c) Hold down loads for pressurized water reactors (which should balance the hydrodynamic lift-off forces and the geometrical changes of the core cavity and of the fuel assemblies under irradiation);

~~(c)~~(d) Temperature gradients;

~~(d)~~(e) Hydraulic forces, including cross-flows between distorted fuel assemblies or mixed fuel assembly concepts;

~~(e)~~(f) Irradiation effects (e.g., irradiation induced growth and swelling);

~~(f)~~(g) Fuel ~~element~~ rod vibration and fretting wear (grid-to-~~fuel element~~ rod fretting for light water reactors, between spacers for pressurized heavy water reactors) induced by coolant flow;

~~(g)~~(h) Creep deformation of the fuel assembly structure (that could lead to fuel assembly distortion);

~~(h)~~(i) Safe shutdown earthquake loading conventionally combined with loss-of-coolant accidents; and

~~(i)~~(j) Postulated initiating events (i.e., anticipated operational occurrences and design basis accidents) and design extension conditions without significant fuel degradation.

~~3.40.~~3.43. For all applicable plant states, the following mechanical safety aspects should be addressed in the design of fuel ~~elements~~ rods and fuel assemblies:

(a) The clearance within and adjacent to the fuel assembly should provide space to allow for irradiation induced growth and bowing (light water reactors) and bulging of the fuel channel (for boiling water reactors, see Annex II, Fuel channel, for supplementary information);

(b) Bowing of fuel ~~elements~~ rods or distortion of assemblies should be limited so that thermalhydraulic behaviour, power distribution, fuel performance and fuel handling are not adversely affected;

(c) F~~f~~atigue ~~strain~~ should not cause the failure of any component of the~~a~~ fuel assembly;

(d) Fuel assembly distortion as a result of mechanical and hydraulic hold down forces and in-core cross-flows should be limited to a level which does not impact the local critical heat flux margins. Also, the fuel assembly distortion should not impair the insertion of the reactivity control cluster assembly (e.g., increase of drop time in pressurized water reactors) to ensure safe reactor shutdown during all applicable plant states (for light water reactors); ~~and~~

(e) Vibration and fretting damage should not affect the overall performances and functions of the fuel assembly and the support structure; and~~.~~

~~(e)~~(f) Hydraulic and mechanical loads (including those resulted from accidents and design basis earthquake) should not cause the failure of any component of the fuel assembly.

~~3.41.~~3.44. For accident conditions (design basis accidents and design extension conditions without significant fuel degradation), the design should prevent any interaction between fuel ~~elements~~ rods or fuel assemblies and fuel assembly support structures that would impede safety systems from performing their functions as specified in the safety analysis. In particular, the following should be assured:

(a) Proper functioning of the components of safety systems (e.g., shutdown devices and their guide tubes for pressurized water reactors); and

(b) Proper cooling of the core.

Fuel pellet-cladding interaction

~~3.42.~~3.45. The design should assure that no fuel cladding failure takes place due to pellet-cladding mechanical interaction during normal operation and anticipated operational occurrences (see Annex II, Pellet-cladding interaction, for supplementary information). Fuel ~~element~~ rod design and plant specific guidelines for power changes during normal operation should ensure that excessive pellet-cladding mechanical interaction is prevented.

During ~~rapid~~ design basis accidents that lead to rapid power transients (e.g., reactivity initiated accident), fuel cladding can ~~be failed~~fail due to excessive pellet-cladding mechanical interaction combined with cladding embrittlement due to in-reactor hydriding at high burnup. Fuel failures corresponding to this failure mode should be considered in safety analysis.

~~3.43.~~3.46. The design should assure that likelihood of stress corrosion cracking in the fuel cladding is minimized during normal operation and anticipated operational occurrences. (For supplementary information, see Annex II, Pellet-cladding interaction.)

Stress corrosion cracking of the fuel cladding should be prevented by implementing adequate design methods such as those given in the following examples:

(a) Reduce tensile stresses in the fuel cladding by restricting rates of power change (allowing for the cladding stresses to relax) or by delaying the time at which the pellet-cladding gap closes (this can be achieved by increasing the initial fill gas pressure in the fuel ~~element~~ rod or by optimizing the creep properties of the cladding);

(b) Reduce corrosive effects of the fission products (e.g., iodine, cadmium, caesium) generated by the pellet by using a liner (for boiling water reactors) or a graphite coating (for pressurized heavy water reactors) that is less susceptible to the corrosive effects on the inner surface of the cladding. This liner can also smooth down the stress concentration;

(c) Reduce the availability of corrosive fission products at the pellet-cladding interface by using additive fuels which are able to better retain the corrosive fission gas products within the fuel matrix; and

(d) Reduce local power peaking factors (and thus changes in local linear heat generation rates ~~changes~~) through core design techniques.

~~3.44.~~3.47. The power-ramp failure threshold should be established, if applicable, in test reactors by means of ~~in-pile~~ power ramp tests, for each type of fuel or cladding. The database should cover the entire burnup range (see Annex II, Pellet-cladding interaction, for supplementary information).

Fuel performance analysis codes can be used to analyze and interpret the power-ramp database and define a failure threshold. ~~, where the~~The evaluation parameter used to define this threshold is usually the maximum cladding stress but strain energy density can also be used. These same fuel performance analysis codes can be used to assess risk factors that cause this type of stress corrosion cracking of fuel ~~elements~~ rods in the ~~commercial~~ reactor core and to define adequate guidelines.

Fuel design limits

~~3.45.~~3.48. Fuel design limits should be established based on all physical, chemical and mechanical phenomena that affect the performance of fuel ~~elements~~ rods and fuel assemblies for all applicable plant states.

Design limits for operational states

~~3.46.~~3.49. For normal operation and for anticipated operational occurrences, the design of fuel ~~elements~~ rods should address throughout whole reloading cycle at least the following ~~limits~~limitations:

(a) No melting occurs in any location within the fuel pellets;

(b) No cladding overheating occurs (e.g., no departure from nucleate boiling in pressurized water reactors, critical power ratio below limits in boiling water reactors, no dryout conditi~~ons~~I for pressurized heavy water reactors);

(c) Fuel cladding does not collapse (light water reactor fuel only);

(d) Rod internal pressure does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between fuel pellets and the coolant (fuel pellet-cladding gap reopening by cladding lift-off);

(e) Fuel cladding corrosion and hydriding do not exceed specified allowable limits; ~~and~~

(f) Cladding stress and strain remain below specified allowable limits; and~~.~~

~~(f)~~(g) Cladding wall thickness reduction (e.g., wear, erosion) does not exceed specified allowable limits.

~~3.47.~~3.50. Components of fuel ~~elements~~ rods and fuel assemblies for light water reactors should be designed to maintain low deformation and growth so that

(a) No geometrical interaction between fuel ~~elements~~ rods and fuel assembly top and bottom nozzles occurs (~~(~~in order to avoid fuel ~~elements~~ rods and fuel assemblies bow in light water reactors). No geometrical interaction between the fuel bundle string and the shield plugs occurs (in pressurized heavy water reactors);

(b) No abnormal local power peaking occurs in the fuel ~~elements~~rods;

(c) No degradation of the critical heat flux performance of the fuel assembly occurs;

(d) Reactor scram or other movement of control rods is not impeded; and

(e) Handling of fuel assemblies is not hampered.

~~3.48.~~3.51. To prevent ~~maintain low probability of~~ fuel cladding failures ~~occurrences~~ caused by pellet-cladding mechanical interaction, possibly assisted by stress corrosion cracking, appropriate operating limits on power changes and power ramp rates of change should be determined such that the power-ramp failure thresholds are not exceeded.

~~3.49.~~3.52. The fuel assembly, other reactor vessel internals and the reactor cooling system should be designed to minimize the risk of any obstruction of the coolant flow due to the release of loose parts or debris, so as to prevent fuel damage in operational states.

~~3.50.~~3.53. Discharge burnup limits, which depend on the fuel ~~elements~~ rods and fuel assembly performance and on the fuel management approach, should be assessed and justified accordingly.

Design limits for design basis accidents and design extension conditions without significant fuel degradation

~~3.51.~~3.54. For design basis accidents and design extension conditions without significant fuel degradation, fuel ~~elements~~rod design should be such that:

(a) The number of fuel ~~element~~ rod failures does not exceed a ~~certain~~ small percentage of the total number of fuel ~~elements~~ rods in the reactor core to ~~limit~~ minimize the radiological consequences of each accident under consideration~~; to within the onsite and offsite release limitations;~~

(b) In determining the total number of fuel ~~element~~ rod failures, all known potential failure mechanisms are evaluated. Chemical reactions including oxidation and hydriding, cladding ballooning or collapse of the cladding, or damage to the cladding caused by an increase in the fuel enthalp~~y a~~Ire some of the failure mechanisms to be considered;

(c) Limits employed in assessing the risk for loss of cladding integrity are based on experimental studies. In determining the limits, chemical, physical, hydraulic and mechanical factors affecting the failure mechanisms as well as the dimensional tolerances of the fuel ~~elements~~rods, are comprehensively and conservatively evaluated. When fuel failure mechanisms and fuel failure limits are burnup dependent, irradiation effects on cladding and fuel properties should be considered in the experimentation and incorporated in the analyses to ensure the application of the experimental results is comprehensive; and

(d) Fuel failure is considered to occur if the radial average enthalpy of a fuel ~~element~~ rod at any axial location, calculated with validated tools, exceeds a certain value to be determined based on representative experimental test~~s~~ results by appropriately adjusting test conditions to represent in-reactor conditions (test parameters to be accounted for include: coolant temperature, coolant pressure, coolant flow rate, reactivity insertion kinetics and ~~element~~ fuel rod internal pressure). Since cladding mechanical resistance changes with irradiation and may vary from one cladding type to another, the reactivity initiated accident failure limit is expected to be dependent on the fuel rod burnup and on the cladding material.

~~3.52.~~3.55. ~~Fuel~~ C~~c~~ore coolability should not be endangered due to, for example:~~,~~

(a) Excessive ballooning or bursting of the fuel ~~elements~~ rods (e.g., in a loss-of-coolant accident event);

(b) Significant deformation of fuel assembly components or reactor internals (e.g., in a seismic event); and

(c) Flow blockage or other consequences due to fuel dispersal and fuel coolant interaction as a result of fuel cladding failure (e.g., in a reactivity initiated accident event).

The design ~~for~~ of fuel ~~elements~~ rods should also be adequate to prevent undesired consequences of reactivity initiated accidents that may cause damage to the reactor coolant pressure boundary or damage to impair the capability to cool the core. This is generally ensured by means of limits on the maximum fuel enthalpy and on the allowable increase in fuel enthalpy.

~~3.53.~~3.56. To ensure the structural integrity of the fuel ~~elements~~ rods is preserved, the following design limits should be defined and justified:

(a) Peak cladding temperature during the accident conditions should not exceed a level where cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on peak cladding temperature due to ~~of~~ fuel fragmentation and its axial relocation within the ballooned area of the fuel rod ~~on peak cladding temperature~~ should be assessed as appropriate. Possible effects of fuel particles dispersal on dose consequences and core coolability should also be addressed;

(b) Total cladding oxidation should remain below limits such that the cladding can still ~~The cladding should not be oxidized during an accident conditions to such a degree that it cannot~~ withstand accident induced loadings (e.g., loss-of-coolant accident quenching phase). The ~~assessment~~ limits should be determined by experiments takinge into account both pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner-side oxidation), –pre-transient and transient hydrogen absorptions as well as chemical interactions between fuel pellets and cladding material; ~~. The hydrogen absorbed by the cladding during normal operation and during accident conditions should not cause the cladding mechanical properties to deteriorate excessively. The effect of the absorbed hydrogen on cladding resistance and ductility should be determined experimentally;~~

(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values which account for initial fuel ~~element~~ rod conditions (e.g., pre-transient hydrogen content of the cladding and fuel burnup); ~~and~~

~~(c)~~(d) If applicable, fuel centerline melting should be limited to a small fraction of fuel pellet volume; and

~~(d)~~(e) Fuel ~~elements~~ rods should withstand loadings resulting from post-transient fuel assembly handling, ~~interim~~ storage~~,~~ and transport to a reprocessing or~~and long-term storage~~ disposal facility.

~~3.54.~~3.57. For light water reactors, the amount of hydrogen generated by the chemical reaction between the coolant and the cladding during a loss-of-coolant accident should not exceed a fraction (e.g., 1%) of the amount of hydrogen that would be generated assuming all claddings surrounding the fuel pellets in the reactor core (excluding the cladding surrounding the plenum volume) reacted with the coolant.

~~3.55.~~3.58. ~~Dispersal of molten fuel particles~~ ~~I~~In case ~~of~~ fuel cladding failures during ~~a reactivity~~reactivity initiated accident ~~transient~~cannot be prevented, dispersal of molten fuel particles should ~~be prevented~~not challenge core coolability. This can be achieved by assuring that the radial average enthalpy at any axial location of any fuel ~~element~~ rod should not exceed ~~a certain~~ burnup dependent limits ~~value~~ derived from, for example, the analysis of a prototypical experimental database.

~~3.56.~~3.59. Structural deformations of fuel ~~elements~~rods, fuel assemblies, control rods or reactor internals should remain limited so as to avoid any impairment of control rod~~s~~ movements in the reactor. In addition, melting temperatures should not be exceeded in control rods at any time or in any location.

CORE STRUCTURES AND COMPONENTS MECHANICAL DESIGN

Design considerations

~~3.57.~~3.60. The reactor core structures and components (see Annex II, Core components, for clarification of ~~the~~ terminology) should be designed to maintain their structural integrity for all applicable plant states, under various damage mechanisms caused by, for example: vibration (mechanical or flow induced) and fatigue; debris effects; thermal, hydraulic, mechanical loads (e.g., loss-of-coolant accidents, seismic events); chemical~~,~~ ~~hydraulic~~ and irradiation effects (including radiation induced growth). ~~; and seismic motions.~~ Of particular concern are: damage to reactivity control and shutdown devices; and damage to the reactor coolant pressure boundary. The effects of high pressures, high temperatures, temperature variations and the temperature distribution, corrosion, radiation absorption rates and the lifetime radiation exposure on physical dimensions, mechanical loads and material properties should be addressed.

In addition, the design of solid reactivity control devices should also ensure withstanding handling loads during refueling operations, transport and storage ~~in addition to above~~.

Important items that are typically addressed within the design of reactivity control assembly, neutron source assembly and hydraulic plug assembly are described in Annex I.

~~3.58.~~3.61. The design of the support structures in the core should provide adequate safety margins for thermal stresses generated in all applicable plant states and should account for additional effects induced by gamma heating on their cooling and thermal responses. The chemical effects of the coolant and the moderator on these structures, which include corrosion, hydriding, stress corrosion and crud buildup should also be addressed.

~~3.59.~~3.62. Provision for the ~~necessary~~ inspection of the core components and associated structures should be included in the design of the fuel assembly, control rods and guide structures, and support structures.

~~3.60.~~3.63. In light water reactors, the core support structures comprise tube sheets, a core barrel, support keys, which maintain the fuel assembly support structures in the desired geometrical position within the core cavity. These core support structures and fuel assembly support structures should be designed to withstand static and dynamic loads including those induced by refueling and fuel handling.

~~3.61.~~3.64. The structures and guide tubes for the shutdown and reactivity control devices and for instrumentation should be designed so that these devices and instrumentation ~~are accurately located and~~ cannot be moved by inadvertent operator actions, strains on equipment, hydraulic forces due to coolant flow or movements of bulk moderator for ~~each~~ all applicable plant states. The design should facilitate the replacement of these devices and instrumentation ~~whenever necessary~~. The design should consider the possibility that flow induced vibration of these devices, instruments or their guide tubes may result in fretting, wear and consequent failure in long term operation. Dimensional stability of the guide structures over their lifetime should also be addressed in the design.

~~3.62.~~3.65. In the case of shutdown and reactivity control devices immersed in a bulk moderator (e.g., for pressurized heavy water reactors), the design should be able to accommodate the effects of hydraulic forces on these structures.

~~3.63.~~3.66. The design should facilitate the replacement of the reactivity control and shutdown devices ~~whenever necessary~~ without causing damage to other reactor core components, unacceptable insertion of reactivity, or ~~undue~~ excessive personnel radiation exposures.

~~3.64.~~3.67. Depending on the reactor type, various other structures may be installed within the reactor vessel. These include, for example, feedwater spargers, steam separators, steam dryers, core baffles, reflectors, and thermal shields. The functions of these internals include flow distribution for the reactor coolant, separation of steam and moisture, or protection of the reactor vessel from the effects of gamma radiation heating and neutron irradiation. These structures should be designed in accordance with paras 3.60–3.61 so that their mechanical performance does not jeopardize the performance of any reactor core safety functions throughout their service life.

Design limits for mechanical design of core structure and components

~~3.65.~~3.68. The design should meet limits specified in the applicable codes and standards that are selected according to safety classification in paras 2.13–2.15~~2.13-2.15.~~.

REACTOR CORE CONTROL, SHUTDOWN AND MONITORING SYSTEMS

Reactor core control system

~~3.66.~~3.69. This section describes important considerations on the control system for maintaining the shapes, levels and stability of the neutron flux within specified limits in all applicable plant states, in order to fulfill Requirement 45 of ~~Ref.~~ [1]. To achieve the objectives for reactor core control, as stated in Requirement 45, para. 6.4 of ~~Ref.~~ [1], adequate means of detecting the neutron flux distributions in the reactor core and their changes are required for the purpose of ensuring that there are no regions of the core in which specified limits could be exceeded.

~~3.67.~~3.70. The core design should allow for the installation of the necessary instrumentation and detectors for monitoring the core parameters such as the core power (level, distribution and time dependent variation), the conditions and physical properties of the coolant and moderator (flow rate and temperature), and the expected effectiveness of the means of reactor shutdown (e.g., the insertion rate of the absorber devices compared with their insertion limits), so that any necessary corrective action can be taken. The instrumentation should monitor relevant parameters over their expected ranges for all applicable plant states including refueling.

Reactivity control devices

~~3.68.~~3.71. The means of control of reactivity should be designed to enable the power level and the power distribution to be maintained within safe operating limits. This includes compensating for changes in reactivity to keep the process parameters within specified operating limits, such as those associated with:

(a) Normal power manoeuvers;

(b) Changes in xenon concentrations;

(c) Effects relating to temperature coefficients;

(d) Rate of flow of coolant or changes in coolant/~~ or~~ moderator temperature and density;

(e) Depletion of fuel and of burnable absorber; and

(f) Cumulative neutron absorbing by fission products.

~~3.69.~~3.72. Reactivity control devices should be capable of ~~used to~~ maintaining the reactor in a subcritical condition, with consideration given to design basis accidents and their consequences. Provisions should be included in the design to maintain subcriticality for plant states in which normal shutdown, fuel cooling or the integrity of the primary cooling system is temporarily disabled (an example includes the situation when the reactor vessel is open for maintenance or refueling in light water reactors).

The types of reactivity control devices used for regulating the core reactivity and the power distribution for different reactor designs are described in Annex II, Control.

~~3.70.~~3.73. The use of control rods or systems as the means of reactivity control for normal operation should not affect adversely their capability and efficiency required to execute fast reactor shutdown.

~~3.71.~~3.74. The maximum degree of positive reactivity and its rate of increase by insertion in all applicable plant states are required to be either limited and/or compensated to prevent any resultant failure of the pressure boundary of the reactor coolant systems, to maintain the capability for cooling and to prevent any significant damage to the reactor core (Requirement 45, para. 6.6 of Ref. [1]).

~~3.72.~~3.75. The arrangement, grouping, speed of withdrawal and withdrawal sequence of the reactivity control devices, used in conjunction with an interlock system, should be designed to ensure that any ~~credible~~ abnormal withdrawal of the devices, which has to be dealt with in the safety assessment, does not cause the specified fuel limits to be exceeded.

~~3.73.~~3.76. Reactivity control systems using a soluble absorber should be designed to prevent any unanticipated decrease in the concentration of absorber in the core~~, which~~ that could cause specified fuel limits to be exceeded. Those parts of systems that contain soluble absorbers such as boric acid should be designed to prevent precipitation (e.g., by heating of the components) (see ~~Ref.~~ [12]). The concentrations of the soluble absorber in all storage tanks should be monitored. Whenever enriched boron (B-10) is used, appropriate monitoring should be provided.

~~3.74.~~3.77. A detailed functional analysis of the control systems alignments and operational conditions should be performed to identify any potential for inadvertent dilution of boron in operation and in shutdown conditions, and to ensure the adequacy of preventive and recovery measures. Such preventive measures may include: permanent administrative locking (of valves or parts of circuits); active isolation actions; interlocks of external injection systems; monitoring of boron concentrations in connected vessels or piping systems; and interlocks for starting recirculation pumps.

~~3.75.~~3.78. The effectiveness of the reactivity control devices such as neutron absorber rods should be verified by direct measurement.

~~3.76.~~3.79. The design of reactivity control devices, as stated in Requirement 45, para. 6.5 of ~~Ref.~~ [1], is required to account for wear out and for the effects of irradiation, such as burnup, changes in physical properties and production of fission gases.

In particular, the following environmental effects should be addressed in the design of control systems:

(a) Irradiation effects – Depletion of the absorber material or swelling and heating of materials due to neutron and gamma absorption. Control rods should be replaced or exchanged accordingly;

(b) Chemical effects – Chemical effects such as corrosion of the reactivity control devices. The transport of activated corrosion products through the reactor coolant system and moderator system should also be addressed; and

(c) Changes in structural dimensions – Dimensional changes or movements of internal core structures due to temperature changes, irradiation effects or external events such as earthquakes should not prevent the insertion of the reactivity control devices.

Reactor shutdown system

3.80. This section describes important considerations for systems designed to transition the reactor to a subcritical state from all applicable plant states, and to maintain it in this state, according to Requirement 46 of~~ Ref.~~ [1]. Together with this safety requirement, Requirement 61 of~~ Ref.~~ [1] for protection system applies to the reactor shutdown system.

~~3.77.~~3.81. Reactor shutdown system should assure that for all applicable plant states, design limits for shutdown margin (paras 3.18–3.19~~3.18-3.19~~) are not exceeded. The necessary reliability should be ensured through the design of the equipment. In particular, the design should ensure the necessary independence between plant processes, control and protection systems.

~~3.78.~~3.82. As stated in Requirement 46, para.6.7 of~~ Ref.~~ [1], the effectiveness, speed of action and shutdown margin of the means of shutdown of the reactor are required to be such that fuel design limits are met. Guidance on the rate of shutdown is described below in the following three paragraphs.

~~3.79.~~3.83. The rate of shutdown should be adequate to render the reactor subcritical with an adequate margin so that the specified design limits on fuel and on the reactor system pressure boundary are met.

~~3.80.~~3.84. In designing for or evaluating the rate of shutdown, the following factors should be addressed:

(a) Response time of the instrumentation to initiate the shutdown;

(b) Response time of the actuation mechanism of the means of shutdown;

(c) Location of the shutdown devices (depending on ~~for~~ the chosen reactor core designs);

(d) Ease of entry of the shutdown devices into the core. This can be achieved by the use of guide tubes or other structural means to facilitate the insertion of devices including the possible incorporation o~~f f~~Ilexible couplings to reduce rigidity over the length of the devices; and

(e) Insertion speed of the shutdown devices. One or more of the following can be used to deliver the necessary insertion speed:

- Gravity drop of shutdown rods into the core,

- Hydraulic or pneumatic pressure drive of shutdown rods into the core, and
- Hydraulic or pneumatic pressure injection of soluble neutron absorber.

~~3.81.~~3.85. Means of checking the insertion speed of shutdown devices should be provided. The insertion time should be checked regularly (typically ~~, for example,~~ at the beginning of each cycle) and possibly during the cycle if the margins to the limits ~~is~~ are not sufficient.

~~3.82.~~3.86. As stated in Requirement 46, para. 6.8 of ~~Ref.~~ [1], in judging the adequacy of the means of shutdown of the reactor, consideration is required to be given to failures arising anywhere in the plant that could render part of the means of shutdown inoperative (such as failure of a control rod to insert) or that could result in a common cause failure. Generally, the characterization of the failure of a control rod to insert should assume that the most reactive core conditions arise when the shutdown device that has the highest reactivity worth cannot be inserted into the core, i.e., the assumption that one shutdown device is stuck.

Different means of shutdown

~~3.83.~~3.87. As indicated in paras Requirement 46, 6.9-6.10 of ~~Ref.~~ [1], at least two independent and diverse shutdown systems are required to be provided, and at least one of two different shutdown systems is required to be capable, on its own, of maintaining the reactor in a subcritical state with an adequate margin and with high reliability even for the most reactive conditions of the reactor core for anticipated core coolant temperatures.

TABLE 1 provides typical examples that illustrate the diversity of means of shutdown ~~that may be used~~ for three different reactor types (boiling water reactors, pressurized water reactors and pressurized heavy water reactors).

TABLE 1. MEANS OF SHUTDOWN

Reactor type	**Fast shutdown system**	**Diverse shutdown system**
Boiling water reactor	B_4C in steel tubes/Hafnium plates (or a hybrid design)	Boron solution injected into moderator/coolant
Pressurized water reactor	Ag–In–Cd in steel tubes/B_4C in steel tubes, Hafnium rods	Boron solution injected into moderator/coolant

Reactor type	Fast shutdown system	Diverse shutdown system
Pressurized heavy water reactor	-Cadmium elements sandwiched and sealed between stainless steel tubes moving in zirconium alloy guide tubes	Gadolinium solution injected into low pressure moderator (Note 1)

Note 1: This shutdown system can also act as another fast shutdown system.

Reliability

~~3.84.~~3.88. The design should include the following measures to achieve a high reliability of shutdown by means of each the following measures or a combination of these as appropriate:

(a) Adopting systems with uncomplicated design and simple operation, and with automatic activation;

(b) Selecting equipment of proven design;

(c) Using a fail-safe design as far as practicable (see Annex II, Shutdown, for supplemen~~ta~~tary information);

(d) Giving consideration to the possible modes of failure and adopting redundancy in the activation of shutdown systems (e.g., sensors). Provision for diversity may be made, for example, by using two different and indenpendent physical trip parameters for each accident as far as practicable;

(e) Functionally isolating and physically separating the shutdown systems (this includes the separation of control and shutdown functions) as far as practicable, on the assumption of credible modes of failure and including common cause failure;

(f) Ensuring easy entry of the means of shutdown into the core, with consideration of the in-core environmental effects of operational states and accident conditions within the design basis;

(g) Designing to facilitate maintenance, in-service inspection and operational testability;

(h) Providing means for performing comprehensive testing during commissioning and periodic refueling or maintenance outages;

(i) Testing of the actuation mechanism (or of partial rod insertion, if feasible) during operation; and

(j) Designing to function under extreme conditions (e.g., seismic).

~~3.85.~~3.89. The design of shutdown systems, as stated in Requirement 45, para. 6.5 of ~~Ref.~~ [1], is required to account for wear out of the control rod cladding and for the effects of irradiation, such as burnup, changes in physical properties and production of helium gases. ~~The bullet~~ I~~i~~tems (a) – (c) in para. 3.79~~3.79~~ are also applicable to the design of shutdown systems. Specific recommendations for diverse shutdown systems injecting neutron absorbers to the reactor coolant system are described in ~~Ref.~~ [12].

Effectiveness of shutdown ~~and reactivity holddown~~system

~~3.86.~~3.90. As indicated in Requirement 46, para. 6.11~~-of~~ ~~Ref.~~ [1], the means of shutdown is required to be adequate to compensate ~~prevent~~ any foreseeable increase in reactivity leading to unintentional criticality during the shutdown, or during refueling operations or other routine or non-routine operations in the shutdown state. The long term ~~reactivity holddown~~shutdown requirements and deliberate actions that increase reactivity in the shutdown state (e.g., the movement of absorbers for maintenance purposes, the dilution of the boron content and refueling actions) should be identified and evaluated to ensure that the most reactive condition is addressed in the criticality analysis.

~~3.87.~~3.91. The design should determine the number and the reactivity worth of shutdown rods by considering various factors. Important factors to be ~~accounted~~accounted for include:

(a) Core size;

(b) Fuel type and the core loading scheme;

(c) Necessary and required ~~margin~~margin of subcriticality;

(d) Assumptions related to failure of shutdown device(s);

(e) Uncertainties associated with the calculations;

(f) Shutdown device shadowing (see Annex II, Shutdown, for terminology clarification); and

(g) The most reactive core conditions after shutdown. These are the result of a number of parameters such as:

- The most reactive core configuration (and where appropriate the corresponding boron concentration) that will occur during ~~the~~whole ~~intended fuel~~ reloading cycle, including during refueling;
- The most reactive credible combination of fuel and moderator temperatures;
- Amount of positive reactivity insertion resulting in design basis accident conditions;
- Amount of xenon as a function of time after shutdown; and
- Burnup of the absorber.

~~3.88.~~3.92. The effectiveness of shutdown system ~~and reactivity hold-down~~ should be demonstrated ~~via~~:

(a) In design, by means of calculation;

(b) During commissioning and prior to startup after each refueling, by means of appropriate neutronic and process measurements to confirm the calculations for the given core loading; and

(c) During reactor operation, by means of measurements and calculations covering the actual and anticipated reactor core conditions.

These analyses should cover the most reactive core conditions, and include the assumption of the failure of shutdown device(s). In addition, ~~reactivity hold-down~~shutdown margin should be maintained if a single random failure occurs in the shutdown system.

~~3.89.~~3.93. If the operation of the ~~reactivity~~ reactor ~~hold-down~~shutdown system is manual or partly manual, the necessary prerequisites for manual operation should be met (see ~~Ref.~~ [13]).

~~3.90.~~3.94. Part of the means of shutdown may be used for the purposes of reactivity control and flux shaping in normal operation. Such use should not jeopardize the functioning of the shutdown system under any condition in all applicable plant states.

~~3.91.~~3.95. The shutdown systems should be testable, as far as practicable, during operation in order to provide assurance that the systems are available on demand.

Separation of protection systems from control systems

~~3.92.~~3.96. As stated in Requirement 64 of ~~Ref.~~ [1], protection systems are required to be physically and functionally separated from control systems to avoid failures of control systems causing failures in the protection system. Guidance on separation of the protection system from other systems is described in ~~Ref.~~ [13].

Partial trip system

~~3.93.~~3.97. In some reactor designs, when measured core parameters (e.g., temperatures, pressures, levels, flows and flux) exceed certain plant design limits, a partial trip system can be activated for protection of the reactor. If applicable, the design should ensure that a partial trip triggered by any anticipated operational occurrence transient does not allow exceeding specified fuel design limits.

Operating limits and setpoints

Operating limits for control system

~~3.94.~~3.98. The design should include operating limits and associated setpoints for actions, alarms or reactor trip to ensure that the operating power distributions remain within the design power distributions.

Limits and set points should consider impacts of the fuel burnup, shadowing effects and coolant stratification (coolant temperature distribution).

~~3.95.~~3.99. Determination of the operating limits and setpoints should include effects of the ageing of the reactor coolant system (e.g., steam generator tube plugging in pressurized water reactors, increase of the diameter of the pressure tube in pressurized heavy water reactors).

Setpoints for reactor core protection

~~3.96.~~3.100. The setpoints should be established and used to control or shut down the reactor at any time during operation. The automatic initiation of control and protection systems during a reactor transient should prevent damage to the nuclear fuel and, in the early stages of a reactor accident, should minimize the extent of damage to the fuel.

~~3.97.~~3.101. Equipment, performance, operating limits and procedures should be defined to prevent excessive control rod worths or reactivity insertion rates. Their capability should be demonstrated. Where feasible, an alarm should be installed to function when any such limit or restriction is violated or is about to be violated.

~~3.98.~~3.102. The design limits, uncertainties, operating limits, instrument requirements, and setpoints should be ~~translated into~~stated in ~~the~~ technical specifications to be used by ~~for the~~ facility operators.

Core monitoring system

~~3.99.~~3.103. As established by Requirement 59 of ~~Ref.~~ [1], core monitoring instrumentation is required to support reactor protection and control systems, as well as to supply sufficiently detailed and timely information on the local heat generation conditions prevailing in the core. The core design should accommodate the detectors and devices for monitoring the magnitude and changes of core power, as well as the local distribution of heat generation in the core, in order to enable any required modification of core parameters (e.g., control rod insertion position, neutron flux, reactor coolant temperature and pressure) within their defined operating ranges. The rapidity of the variation in a parameter should determine whether the actuation of the reactor control systems is automatic or manual.

In addition, the radionuclide activity levels in the coolant should be monitored to assess the integrity of the fuel system during operation and to verify that design or operational limits are not exceeded.

~~3.100.~~3.104. The core monitoring parameters ~~such as the following examples to be measured~~ should be adequately selected. This selection~~, which~~ will depend on the reactor type. The following are examples of parameters to be measured for the purposes of core monitoring:

(a) Spatial distribution of the neutron flux and related power distribution peaking factors;

~~(a)~~(b) Reactor coolant system pressure;

~~(b)~~(c) ~~(b)~~ Coolant~~ant~~ temperature (e.g., inlet temperature, outlet temperature);

~~(c)~~(d) ~~(c)~~ Reactor coolant pump speed~~flow rate~~;

~~(d)~~(e) ~~(d)~~ Water level (for ~~a boiling~~ light water reactors);

~~(e)~~(f) Radionuclide activity in the coolant (see Annex II, Coolant, for supplementary information);

~~(f)~~(g) Control rod insertion position; and

~~(g)~~(h) Concentration of soluble boron ~~and~~ or B-10 content when enriched boron is used (for a pressurized water reactor).

Other safety related parameters may be derived from the measured ~~parameter~~parameters. Examples include:

(a) Neutron flux doubling time,

(b) Rate of change of neutron flux,

(c) Axial and radial neutron flux imbalances,

(d) Reactivity balance, and

(e) Thermalhydraulic core parameters (e.g., core thermal power, liner heat generation rate, reactor coolant flow rate, the departure from nucleate boiling ratio or the critical power ratio).

~~3.101.~~3.105. The accuracy, speed of response, range and reliability of all monitoring systems should be adequate for performing their intended functions (see ~~Ref.~~ [13]). The monitoring system design should provide for the continuous or adequate periodic testing of these systems.

~~3.102.~~3.106. Guidance on post-accident monitoring is provided in ~~Ref.~~ [13]. If core monitoring is needed in accident conditions, for example, to monitor system temperatures, reactor vessel water level, or reactivity changes, the instrumentation to be used should be qualified to withstand the environmental conditions to be expected during and following the accident.

~~3.103.~~3.107. The spatial power distribution should be monitored by means of ex-core or in-core instrumentation (such as neutron detectors and gamma thermometers). Measurements of the local power at different positions in the core should be performed to ensure that adequate safety margins are maintained considering the impact of the spatial power distribution changes due to core control and/or core burnup effects. The in-core power distribution should be monitored routinely. Detectors should be adequately distributed ~~strategically~~ in the core to detect reliably the local changes in power density. Both ex-core and in-core neutron detectors should be calibrated periodically.

~~3.104.~~3.108. A computerized core monitoring system should be used to ensure that the status of the core is within the operating limits assumed in the safety analysis. Qualification of the system should be ensured wherever it is coupled to a~~-~~ protection system (see ~~Ref.~~ [13]).

3.109. During reactor shutdown, a minimum set of instruments or combination of instruments and neutron sources should be available to monitor neutron flux and heat generation distribution (e.g., using flux detectors with an adequate sensitivity) whenever fuel assemblies are present in the reactor vessel, including fuel loading and start-up phases.

~~At least one means of shutdown should be available to assure core subcriticality under cold conditions.~~

~~3.105.~~3.110. During reactor startup in some reactors, a combination of interlocks on flux monitoring systems and reactivity control devices is used to ensure that the most appropriate monitors are used for particular flux ranges and to avoid undue reactor trips. The design of such interlock systems should be consistent with the design of the reactor protection system.

~~3.106.~~3.111. During reactor startup, and especially during the first startup, the neutron flux is very low relative to that in full power operation, so more sensitive neutron detectors may be needed temporarily to monitor the neutron flux. A neutron source may be necessary to increase the flux to a level that is within the range of the startup neutron flux monitors. The design of the neutron sources should ensure that:

(a) The sources function properly to provide sufficient signals from the neutron flux monitors over their planned lifetime; and

(b) The sources are compatible with the fuel assemblies and the fuel assembly support structures.

CORE MANAGEMENT

Design considerations

~~3.107.~~3.112. The primary objective of core management is to ensure the safe, reliable and optimum use of the nuclear fuel in the reactor, while remaining within operational limits and conditions. ~~design limits imposed by the design of fuel elements, fuel assemblies, thermalhydraulics and neutronics.~~

~~3.108.~~3.113. Fuel cycles should be developed with appropriate means of controlling the core reactivity and the power distribution to address fuel design limits.

While the details of core management depend on the reactor type, in all cases the core management program should provide:

(a) Means to perform core management functions effectively throughout the fuel cycle so as to ensure that core parameters remain within core management design limits. Core management functions include: core design (specification of fuel assembly loading and shuffle patterns to provide optimum fuel burnup and desired fluxes), fuel assembly procurement, reactivity determinations and core performance monitoring; and

(b) Core operating strategies that permit ~~maximum~~ operating flexibility ~~for reactor utilization~~ and ~~optimum~~ good fuel utilization ~~-~~while remaining within core management design limits.

Core design

~~3.109.~~3.114. To achieve the desired core reactivity and power distribution for reactor operation, ~~the core management strategies should provide~~ the operating organization should be provided with the following information:

(a) Loading patterns (including enrichment and configuration of fuel ~~elements~~rods) and orientation of fuel assemblies in each fuel cycle (for- light water reactors);

(b) Schedule for the subsequent unloading and loading of fuel assemblies;

(c) Configurations of reactivity control and shutdown devices; and

(d) Burnable absorbers and other core components to be removed, inserted or adjusted.

~~3.110.~~3.115. Core-reload depletions and reactor physics parameters are provided as input to safety analyses, plant monitoring and protection systems, and operator guidance; therefore, these parameters should be analyzed based on pre-determined plant operational objectives and resultant plans. These reactor physics parameters include: reactor start-up conditions (e.g., critical boron and control rod positions~~)~~, reactor kinetics, fuel temperature coefficients~~ents,~~, moderator temperature coefficients, control rod and bank worths, ~~and~~ power peaking factors, etc.).

Unplanned power maneuvering during flexible operation may alter the power and burnup profile across the core relative to those predictions. As such, core-reload depletions and reactor physics parameters predictions should be continuously or periodically examined and evaluated, using relevant monitoring parameters.

~~3.111.~~3.116. The design of the reactor core should include analyses to demonstrate that the fuel management strategy and the established limitations on operation do not change in any manner that would cause nuclear design limits and hence fuel design limits to be violated throughout ~~the~~ whole reloading cycle. ~~reactor operating cycle or lifetime.~~

~~3.112.~~3.117. In reactor core analyses, multi-dimensional and multi-scale physics codes and system thermalhydraulic codes are preferentially used for realistic analysis of the reactor core for all applicable plant states. Uncertainties should be adequately incorporated in the analyses (see [5] for details).

~~3.113.~~3.118. The reactor core analysis should be performed based on typical cases covering ~~the~~ whole reloading ~~entire operating~~ cycle for the following reactor core conditions such as~~h as~~:

(a) Full power, including representative power distributions;

(b) Load following (as applicable);

(c) Approach to criticality and power operation;

(d) Power cycling;

(e) Startup;

(f) Refueling;

(g) Shutdown;

(h) Anticipated operational occurrences; and

(i) Operation at the thermalhydraulic stability boundary (for boiling water reactors).

Whenever the management of fuel in the core is changed or any characteristics of the fuel ~~elements~~ rods (such as the fuel enrichment, fuel ~~element~~ rod dimensions, fuel rod configuration, or the fuel cladding material) are changed, a new core analysis should be performed and documented.

~~3.114.~~3.119. The reactor core analysis should include fuel ~~element~~ rod performance analyses based on average and local power levels and axial temperature distributions to demonstrate that the respective thermal and mechanical fuel design limits are met for all operational states. For light water reactors, the reactor core analysis should include peak channel power and peak linear power rates for normal full power operation and steady state radial and axial power distributions in ~~at~~ each ~~assembly location and axially along the~~ fuel assembly~~(s)~~. Allowance should be made to account for the effects of changes in the geometry of the assembly on neutronic and thermalhydraulic performance (e.g., changes in the moderator gap thickness due to bowing of the assembli~~y~~es). The reactor core analysis should also include the radial power distribution within a fuel assembly and the axial power distortion due to spacers, grids and other components in order to identify hot spots and to evaluate the local power levels.

Refueling

~~3.115.~~3.120. For on-power refueling in pressurized heavy water reactors, the effects of the refueling operation on the neutronic behavior of the core should be demonstrated to remain within the control capability of the reactor control systems.

~~3.116.~~3.121. Safety assessment should address any event that could cause inadvertent criticality during core loading or unloading and during handling phases.

~~3.117.~~3.122. The fuel loading sequence should be monitored through the use of in-core (for boiling water reactors) or ex-core flux distribution measurements, or of special organizational measures. The fuel reload~~ing~~ pattern after reloading should be validated through the use of in-core flux distribution measurements.

~~3.118.~~3.123. The light water reactor core should be designed such that the consequences of the worst misloaded fuel assembly, if any, remain within nuclear and fuel design limits. If a misloaded fuel assembly can be prevented by special measures and equipment, the effectiveness and reliability of these precautionary measures should be demonstrated. Computational analyses should be performed if it cannot be demonstrated that the specified precautionary measures are sufficient.

Core management design limits

~~3.119.~~3.124. The reactor core analysis should verify that the core fuel loading pattern will meet fuel design limits for all applicable plant states.

~~3.120.~~3.125. For practical reasons and simplicity, for light water reactors, a system that develops and monitors the nuclear key safety parameters (refer to para. 3.104) can be used to verify the suitability of the reload core design.

Special core configurations

Mixed core

~~3.121.~~3.126. When fuel assemblies of different types are loaded into the core (~~a~~ so-called mixed core), ~~the~~ all fuel assembly types ~~in the mixed core~~ should meet the fuel design limits for all applicable plant states. The assessment should be performed for ~~be assessed in such a manner to demonstrate that the mixed core meets nuclear design limits for both the~~ the initial and subsequent ~~reload~~ mixed core reloads. ~~, and that the fuel assemblies in the mixed core meet fuel design limits for all applicable plant states. These assessments~~It should include: the dimensional, mechanical and thermalhydraulic response of the various fuel types (e.g., in terms of ~~the~~ pressure drop characteristics through the fuel assembly(s) and flow rate), the compatibility ~~–~~with ~~the~~neutronic and thermalhydraulic ~~–nuclear~~

characteristics of the original core and with the related safety analyse~~i~~s. The critical heat flux or critical power correlation used in the core monitoring system should be valid for all fuel assembly types present in the mixed core.

~~3.122.~~3.127. Relevant nuclear parameters such as reactivity, reactivity coefficients, control rod worth and power distributions should be evaluated for the different fuel assembly designs. The compatibility evaluation may be developed based on single fuel assembly calculations in an infinite medium. The combined effects on the related core-wide parameters should be evaluated.

Mixed-oxide fuel core

~~3.123.~~3.128. The design of a mixed-oxide core should include analyses to ensure that nuclear design limits (for both initial and subsequent reload cores) and fuel design limits are met for all applicable plant states. In the analyses, following considerations should be addressed:

(a) The mixed-oxide fuel properties (see Annex II, Fuel, for supplementary information) are somewhat different from the UO_2 fuel and this should be incorporated in computer codes and models used for the fuel design and safety analyses;

(b) In the mixed-oxide core, control rod and absorber worth are reduced as a result of neutron spectrum hardening due to the higher thermal absorption cross sections of plutonium compared with uranium, and as a result, the reactor shut-down margin can be reduced. To compensate for the reduced shutdown margin, additional control rods ~~-~~or absorption capability of the absorbing materials (e.g., B-10 enrichment increase) should be implemented;

(c) The kinetic parameters for mixed-oxide fuel, namely, the total fraction of delayed neutrons and the prompt neutron lifetime are lower than those for UO_2 fuel. The lower delayed neutron fraction of mixed-oxide fuel can result in a prompt critical reactor condition with a smaller reactivity insertion; thus, there is less time for control rod insertion or boron system injection to provide ~~reactor~~ reactivity control. This should be addressed in the core design and safety analyses for all applicable plant states (e.g., reactivity initiated events as anticipated operational occurrences and design basis accidents~~ transients~~); and

(d) The fission cross sections in mixed-oxide fuel are larger than those in UO_2 fuel, and this can result in steep flux gradients between adjacent mixed-oxide and UO_2 fuel ~~elements~~rods. This effect can be reduced with variations of the plutonium content and core design pattern adjustments. Another consequence of the differences in cross sections between plutonium and uranium are the changes in the moderator temperature coefficient, the fuel temperature coefficient, and the coefficient of reactivity for coolant voids. Core design and safety analyses should evaluate the effects of these changes in reactivity coefficients.

Load Following and Power Manoeuvering

~~3.124.~~3.129. The effects of operating conditions such as load following (see Annex II, Load following, for supplementary information), power cycling, reactor startup, and refueling manoeuverings~~s~~ should, whenever specified~~necessary~~, be superimposed onto the power level distributions and temperature histories to evaluate the potential effects of thermal cycling on fuel ~~element~~ rod thermal mechanical responses, such as the buildup of pressure due to fission gas release to the pellet-cladding gap and fuel cladding fatigue.

~~3.125.~~3.130. Once the extent of the desired flexibility is determined, in-depth evaluation of impacts on the nuclear power plants design and operation (i.e., requirements on the safety analysis and the operational limits and conditions) should be performed. Based on this evaluation, additional specifications for qualification and implementation can be developed.

~~3.126.~~3.131. To assure the control of core reactivity with load following and power manoeuvering, the core and generator power balance and the reactor stability should be maintained.

~~3.127.~~3.132. The operational limits should be adjusted to cover perturbations due to load following operation (see Annex II, Load following, for supplementary information).

Reactor operation with leaking fuel ~~elements~~ rods

3.133. Fuel rod failures can affect ease of access, work scheduling and worker dose for plant operations personnel. ~~.~~Reactor core operation with defective fuel ~~elements~~ rods should stay within the radiochemical requirements (see Annex II, Defective fuel, for supplementary information) as defined by the limit on coolant radionuclide activity included in the Technical Specifications document.

~~3.128.~~3.134. The core design and operations program should establish procedures and limits for operating the core with defective fuel assemblies while assuring radioactive dose limits are not exceeded for plant personnel. In light water reactors, shutdown should be done if the operating radiochemical limits are exceeded, and all defective fuel assemblies are replaced according to procedures after the outage. In ~~boiling water reactors and~~ pressurized heavy water reactors, fission product release from defective fuel and subsequent secondary hydriding of the cladding can be minimized by reducing the power level of defective fuel rods. (See Annex II, Defective fuel, for supplementary information.)

Core re-design after fuel assembly repair

~~3.129.~~3.135. In light water reactors, the fuel assemblies containing damaged and leaking fuel rods may be repaired and reconstituted with replacement rods, dummy rods or vacancies. The use of vacancies should be limited so that design limits are met.

~~3.130.~~3.136. The impact of reconstituted fuel assembly on the design of the reactor core should be assessed.

Impact of fuel design and core management on fuel handling, shipment, storage, reprocessing and disposal

~~3.131.~~3.137. Design limits are determined, based on the concept of defence in depth, to fulfill safety requirements for all applicable plant states. Fuel design limits described in paras 3.49–3.59~~3.49-3.59~~ should be extended to assure that the fuel ~~elements~~ rods and fuel assemblies remain intact (when applicable) or do not degrade further (in case of leaking fuel rods) in the back-end phases after the assemblies are discharged from the core. Back-up phases include: handling, shipment, storage, reprocessing and disposal. The following key in-reactor safety parameters are among th~~ose th~~at may have an impact on the post irradiation behavior of the fuel ~~elements~~ rods and the fuel assemblies:

(a) End-of-life fuel rod internal pressure

Even though fuel ~~elements~~ rods can withstand some extent of over-pressurization exceeding the normal coolant pressure without failure in normal operation, such highly pressurized used fuel ~~elements~~ rods may not be acceptable to handle when coolant counter-pressure is diminished (e.g., in spent fuel storage facilities). This is particularly relevant for mixed-oxide fuels which remains h~~-at h~~igher temperature for a longer period of time and continue to release helium gases from the fuel material.

(b) Massive cladding hydriding and cladding mechanical properties

Localized hydriding (e.g., due to corrosion layer spalling or due to axial pellet-pellet gaps)~~,~~ may not~~t impact~~ hydride normal operation or be of consequence in accident conditions, but such a condition may lead to delayed hydride~~d~~ cracking of zirconium-based alloy cladding ~~ng cracking~~ in post-irradiation handling or storage, or undesired failures in the event of a shipment accident.

(c) Grid-to-~~fuel~~ ~~element~~ rod fretting wear

Localized wear is usually undetected unless it wears through the complete cladding wall thickness and creates a leakage pathway. Some fuel ~~elements~~ rods affected by excessive wear may exhibit localized weakness ~~weakness~~ that may lead to long term creep failures or other mechanical failures in the event of a shipment accident~~s~~.

(d) Discharge burnup

Fuel design, core management and the resultant discharge burnup affect the fuel isotopic vector ~~degradation~~, which in turn will impact the economy of fuel reprocessing or disposal. High discharge burnups degrade ~~spent~~ fuel isotopic compositions and therefore its ~~energetic quality.~~reactivity. In mixed-oxide fuel, Pu content should be adjusted to maintain parity with UO_2 fuel reactivity, up to the anticipated discharge burnup. ~~As a result Pu content in mixed-oxide fuel should be increased to maintain parity with UO_2 enrichment.~~

(e) Others

~~For~~ ~~n~~New fuel ~~element~~ rod or new fuel assembly designs, proposed by the fuel vendors to address other in-reactor issues (e.g., stress corrosion cracking of fuel cladding, fission gas release, ~~and~~ fuel assembly distortion, fuel performance in accident conditions, etc.), ~~-~~should remain compatible with back-end related requirements (back-end phases include handling, shipment, storage, disposal or reprocessing if applicable).~~industrial requirements for reprocessing.~~

4. QUALIFICATION AND TESTING

GENERAL

4.1. Safe operation of the reactor core throughout the lifetime of the structures, systems and components of the reactor core, including the fuel ~~elements~~ rods and assemblies, core components, and control systems requires a robust program for qualification, inspection, and testing of the equipment design and analysis process. This can be achieved as described below.

DESIGN QUALIFICATION

4.2. A qualification program should confirm the capability of the reactor core structures, systems and components to perform ~~its~~ their function, for the relevant time period, with account taken of the appropriate functional and safety considerations under prescribed environmental conditions (e.g., conditions of pressure, temperature, radiation levels, mechanical loading and vibration). These environmental conditions should include the variations expected in normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation.

4.3. The characteristics of certain postulated initiating events may preclude the performance of realistic commissioning tests and recurrent tests that could confirm that structures, systems and components would perform their intended safety functions when called upon to do so, for example in case of ~~an~~ earthquake. For the structures, systems and components concerned and the events considered, a suitable qualification program should be planned and performed prior to their installation.

4.4. Methods of qualification should include:

(a) Performance of a type test on the structures, systems and components representative~~representative~~ of that to be supplied;

(b) Performance of a test on the structures, systems and ~~components~~components supplied;

(c) Use of pertinent past experience;

(d) Analysis based on available and applicable test data; and

(e) Any combination of the above methods.

4.5. Design qualification may be established through operating experience with fuel systems of the same or similar design. The basis for the previous experience should be identified and the performance record should be evaluated. The maximum burnup and core power operating experience should be referenced and the fuel assembly performance should be compared against design criteria identified for phenomena such as fretting wear, oxidation, hydriding~~, and~~ crud buildup, fuel assembly bow, etc.

INSPECTION

4.6. A system should be designed to allow the identification of each fuel assembly and to assure its proper orientation within the core. Following initial core fuel loading or any reload core loading, the locations and orientation of each fuel assembly should be inspected to verify correct location and positioning.

TESTING INCLUDING PROTOTYPE AND LEAD USE ASSEMBLIES

4.7. Provisions should be made in the design for in-service testing and inspection to ensure that the core and associated structures and the reactivity control and shutdown systems will perform their intended functions throughout their lifetime. Further guidance on in-service inspections is provided in ~~Ref.~~ [14].

4.8. Out-of-reactor tests on fuel assembly prototypes should be performed, when practical, to determine the characteristics of a new design. The following out-of-reactor tests are generally performed for this purpose:

Light water reactors

(a) Spacer grid ~~structural~~ tests (including pressure drop tests, crush strength tests and other structural tests such as seismic resistance tests);

(b) Control rod structural and performance tests;

(c) Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)~~.~~;

(d) Fuel assembly hydraulic flow tests, including pressure drop and fuel assembly lift-off force determination, control rod vibration and wear, fuel assembly vibration, grid-to-~~fuel~~-rod-fretting (accounting for spacer grid spring relaxation), and assembly wear and lifetime evaluations; and~~.~~

(e) Fuel assembly thermalhydraulic tests including critical heat flux correlation determination.

Pressurized heavy water reactors

(a) Fuel bundle string~~ltring~~ pressure drop tests;

(b) Cross-flow endurance tests;

(c) Mechanical endurance tests;

(d) Bundle impact tests;

(e) Bundle strength tests;

(f) Wear tests;

(g) Seismic qualification tests;

(h) Wash-in and wash-out tests (where applicable); and

(i) Critical heat flux tests.

4.9. In-reactor testing of design features through irradiations in materials test reactors or through lead-use assembly irradiation should be used to justify the specified maximum burnup or fluence limit for a new design. The f~~fo~~ollowing phenomena may be tested in this manner:

(a) Fuel and burnable absorber rod growth;

(b) Fuel ~~element~~ rod bowing;

(c) Fuel ~~element~~rod, spacer grid, and fuel channel (~~if present~~present) oxidation and hydride levels;

(d) Fuel ~~element~~ rod fretting, and spacer (for pressurized heavy water reactors) fretting

(e) Fuel assembly growth;

(f) Fuel assembly bowing;

(g) Fuel channel (for boiling water reactors) wear and distortion;

(h) Fuel ~~element~~ rod ridging, i.e., pellet-cladding interaction;

(i) Fuel ~~element~~ rod integrity;

(j) Holddown spring relaxation (for pressurized water reactors);

(k) Spacer grid spring relaxation (for light water reactors); and

(l) Control rod and g~~G~~uide tube wear ~~characteristics~~ (for pressurized water reactors).

4.10. In cases where in-reactor testing of a new fuel assembly design or a new design feature cannot be performed, special attention should be given to analytical evaluations and to augmented inspection or surveillance plans to validate the fuel design capability and performance features.

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ANNEX I: ITEMS TO BE ADDRESSED WITHIN THE DESIGN OF THE FUEL ~~ELEMENT~~ROD, FUEL ASSEMBLY, REACTIVITY CONTROL ASSEMBLY, NEUTRON SOURCE ASSEMBLY AND HYDRAULIC PLUG ASSEMBLY

FUEL ~~ELEMENT~~ROD

I-1. The design of fuel ~~elements~~ rods needs to address the following items:

Cladding

(a) Fuel ~~element~~ rod vibration and wear (grid-to-~~fuel~~rod ~~element~~ fretting wear);

(b) Cladding mechanical properties evolution with irradiation (displacement and pressure driven loadings);

(c) Materials and chemical evaluation;

(d) Stress corrosion;

(e) Cycling and fatigue; and

(f) Geometrical and chemical stability of the cladding under irradiation.

Fuel material (including burnable~~ble~~ absorbers)

(a) Dimensional stability of the fuel under irradiation conditions;

(b) Fuel densification (kinetics and amplitude);

(c) ~~Potential~~Potential for chemical interaction with the cladding and the coolant;

(d) Fission gas generation and distribution within the fuel pellets;

(e) Fission gas release kinetics;

(f) Gaseous swelling;

(g) Thermal mechanical properties under irradiation; and

(h) Microstructure changes as a function of irradiation.

Fuel ~~element~~ rod performance

(a) Pellet and cladding ~~temperatures~~temperatures and temperature~~s~~ distributions;

(b) Fuel-clad gap closure kinetics and amplitude (to address pellet-cladding interaction issue);

(c) Irradiation effects on fuel rod behaviour (e.g., fuel restructuring, fuel pellets cracking patterns, solid and gaseous fission product ~~swelling~~swelling, fission gas release and rod internal pressure increases, fuel rod thermal conductivity degradation);

(d) Fuel ~~element~~ rod bowing; and

(e) Fuel ~~element~~ rod growth.

Fuel ~~element~~ rod performance is demonstrated using validated analytical models and/or representative experimental data collected either in test programs or from commercial power plants (lead test~~s~~ fuel rods or lead test fuel assemblies). The models are generally burnup dependent.

FUEL ASSEMBLY

I-2. Fuel assembly components (i.e., top and bottom nozzles, guides tubes, spacers, mixing grids, grid springs, connections and fuel assembly holddown system~~fuel assembly holddown system~~) need to be designed to withstand ~~to withstand~~ the following conditions and ~~loadings~~loads:

(a) Core restraint system ~~loadings~~loads;

(b) ~~Hydrodynamic~~Hydrodynamic ~~loadings~~loads;

(c) Thermalhydraulic limits (e.g., critical heat flux);

(c) Accident loads ~~ings~~ (e.g., ~~seismic,~~ loss-of-coolant accident) and seismic loads;

(d) Handling and shipping ~~loadings~~loads; and

(e) Fuel assembly bow.

REACTIVITY CONTROL ASSEMBLY

I-3. The design of the rod cluster control assembly needs to address the following items:

(a) ~~Interna~~Rod internall pressure and related cladding stresses during normal, transient, and accident conditions;

(b) Thermal expansion and irradiation induced swelling;

(c) Evolution under irradiation of absorber materials and the cladding; and

(d) Fretting wear effect on cladding resistance.

NEUTRON SOURCE ASSEMBLY

I-4. The design of the neutron source assembly needs to address the following items:

(a) Irradiation effects;

(b) Efficiency to account for burnup shadowing effects of peripheral fuel assemblies; and

(c) External events such as earthquakes.

HYDRAULIC PLUG ASSEMBLY

I-5. The design of the hydraulic plug assembly needs to address the following items:

(a) Interaction with guide tubes due to thermal expansion or irradiation induced swelling;

(b) Impact on coolant by-pass flow (for pressurized water reactors); and

(c) Fretting wear effect on guide tube resistance.

ANNEX II: SUPPLEMENTARY TECHNICIAL INFORMATION

II-1. In this Annex, supplementary technical information is provided to clarify the meaning of terminologies that are not defined in the IAEA Safety Glossary [9] but used in this Safety Guide, ~~and~~ or to provide additional background or supporting examples for specified design recommendation paragraphs in this Safety Guide~~s~~.

II-2. Supplementary information provided includes:

Topics	Clarification	Supplement to paragraph no.
Burnable absorber	For light water reactors, in order to maintain a negative moderator temperature coefficient, the designer could choose fixed burnable absorber added to the fuel pellet or fuel assembly in the form of burnable absorber rods to reduce the required concentration of the burnable absorber in the moderator. ~~by adding fixed burnable absorber to the fuel pellet or to the fuel assembly in the form of burnable absorber rods.~~ Burnable absorber could also be used to flatten the power distribution and to reduce variations in reactivity during fuel burnup.	3.19
Cladding	“Cladding leaktightness” is required to prevent the release of volatile fission products, and “cladding structural integrity” is required to prevent the release of solid ~~fission products~~fuel particles to coolant.	2.4
	Zirconium-based alloy materials (e.g., Zircaloy-2, Zircaloy-4, ZIRLO™ and Optimized ZIRLO™, M5®, E110) are typically used for the cladding material. Other innovative cladding materials are under development for use in application such as enhanced accident tolerant fuel, with focus on more benign steam reaction and lower hydrogen generation	3.5
	In-reactor corrosion reduces the bearing thickness of the cladding but the hydriding of the cladding, which is a consequence of the corrosion mechanism, can be more detrimental because it degrades the mechanical properties of the cladding. As a result, some fuel	3.37

Topics	Clarification	Supplement to paragraph no.
	design limits, such as those for reactivity initiated accident and loss of coolant accident are now expressed as a function of the cladding pre-transient hydrogen content rather than the amount of corrosion or the burnup levels.	
	Corrosion and hydriding depend strongly on the material performances and on operating conditions, such as temperature, coolant chemistry and linear heat generation rates (governing, for a given discharge burnup, the allowable irradiation time). These environmental conditions need to be considered. In order not to degrade the corrosion performance of the materials, appropriate water chemistry needs to be implemented (e.g., by maintaining a low oxygen content and the appropriate pH level).	3.38
Control	The types of reactivity control devices used for regulating the core reactivity and the power distribution for different reactor designs include the following: (a) Pressurized water reactor • Use of solid neutron absorber rods; • Use of soluble absorber in the moderator or coolant; • Use of fuel with distributed or discrete burnable absorber; and • Use of a batch refueling and loading pattern. (b) Boiling water reactor • Use of solid neutron absorber blades; • Control of the coolant flow (moderator density); • Use of fuel with distributed or discrete burnable absorber; and • Use of a batch refueling and loading pattern. (c) Pressurized heavy water reactor	3.17, 3.72

Topics	Clarification	Supplement to paragraph no.
	• Use of solid neutron absorber rods; • Use of soluble absorber in the moderator; • Control of the moderator temperature; • Control of the moderator height (for older pressure tube type pressurized heavy water reactors); • Use of liquid absorber in tubes; and • Use of on-power refueling.	
Coolant	Chemical additives to the coolant (e.g., the boric acid used in pressurized water reactors) could be used as neutron absorbers to provide a second system of control over the core reactivity. Other chemical additives (e.g., Zn, H, Li, Cu) can also be used to control the chemistry of the coolant (e.g. to control pH and oxygen content) in order to inhibit corrosion of or crack propagation in core components and reactor internals, and thereby to reduce contamination of the reactor coolant system through crud generation.	3.6
	Reactor coolant activity is measured by the device belonging to the primary coolant makeup and water cleaning system; for details see ~~Ref.~~ [12].	3.104 (f)
Core components	In the IAEA Safety Glossary [9], “core components” ~~are~~ ~~referred~~refer to the elements of a reactor core, other than fuel assemblies, that are used to provide structural support of the core construction, or the tools, devices or other items that are inserted into the reactor core for core monitoring, flow control or other technological purposes and are treated as core elements. Examples of core components are reactivity control devices or shutdown devices, neutron sources, dummy fuel, fuel channels, instrumentation, flow restrictors and burnable absorbers.	3.60

Topics	Clarification	Supplement to paragraph no.
Defective fuel	The iodine spiking phenomenon after plant transients has received particular attention in safety evaluations. For particular pre-accident conditions, its occurrence may increase the radiological consequences of the postulated accident. One approach is to specify a limit to the amount of iodine activity allowed in the reactor coolant after plant transients. Behaviour of leaking fuel ~~elements~~ rods during design basis accidents, e.g., loss-of-coolant accidents, reactivity initiated accidents and steam generator tube rupture, may be specific and may need to be assessed individually. For example, loss-of-coolant accident margins ~~are~~ may not be affected by the presence of leaking fuel because conservative assumptions are specified as requirements for radiological consequence evaluation. Reactivity initiated accident design limits may not be affected ~~need not be affected in case of~~ by the presence of a limited number of leaking fuel rod(s), although it is recognized that a leaking fuel rod has~~s~~ lower capability in withstanding reactivity initiated accident loadings and~~–~~ consequently have a higher probability to cause limited fuel coolant interaction.~~,~~	3.133
	In boiling water reactors, it is often possible to locate the region or regions in the core that contain defective fuel by using the flux tilting method. Once those regions have been identified, it is possible to reduce the power of fuel assemblies in those regions through selective placement of control blades. In pressurized heavy water reactors, defective fuel assemblies can be detected and located by means of tracing fission products elements and delayed neutrons. With a reduced operating power level, the reactor operation with defective fuel core could be continued without significant iodine spikes until defective fuel assemblies are discharged from the reactor. The mitigation effectiveness provided by power suppression is greatest when applied to relatively small, tight defects. For this reason, detection and suppression needs to be undertaken as early as	3.134

Topics	Clarification	Supplement to paragraph no.
	possible once the presence of a leaking fuel ~~element~~ rod is indicated in a core.	
Fuel	“Fuel” means fuel matrix, ~~elements~~ rods and/or assemblies unless otherwise specified.	1.4
	The fuel ~~element~~ rod is interchangeably called the fuel ~~rod~~ element or the fuel pin.	1.4
	The fuel assembly is called the fuel bundle for pressurized heavy water reactors.	1.4
	Innovative fuel materials, e.g., enhanced accident tolerant fuel, with focus on more benign steam reaction and lower hydrogen generation, are under development.	1.6
	“Fuel matrix” refers to the structure/microstructure of various types of ceramic fuel pellets.	2.4
	Examples of the fuel pellet material include: (a) Enriched uranium dioxide (UO_2); (b) Natural uranium dioxide (UO_2) (~~for use~~for use in pressurized heavy water reactors); (c) Mixed oxide (UO_2-PuO_2); (d) Thorium-based fuel (e.g., ThO_2, thorium-blended UO_2, thorium-blended mixed-oxide fuel); (e) Reprocessed uranium dioxide (UO_2); and (f) Doped fuel pellets (e.g., Cr, Al, Si) to improve their performance (for use in light water reactors). Burnable absorber material (e.g., Gd, Dy, B and Er) may be used, for example, blended in sintered UO_2 pellets or coated on their surface, to suppress temporarily the excess reactivity resulting from a high concentration of the fissile material in the fuel.	3.4
	The hot ~~element~~ rod ~~is referred~~refers to the fuel ~~element~~ rod with the highest relative power, considering the conservative radial core power distribution.	3.24 (a)

Topics	**Clarification**	**Supplement to paragraph no.**
	In the assessment of the peak fuel temperatures during operational states, the following burnup dependent phenomena need to be addressed: changes in fuel thermal conductivity/diffusivity and in pellet-cladding gap thermal conductance, fuel densification, fuel swelling, accumulation of fission products in the fuel pellets, fission gas release in the free volumes of the fuel ~~elements~~ rods and any other changes in the pellet microstructure. Due to irradiation effects, the fuel melting temperature varies as a function of fuel burnup and thus needs to be determined using representative irradiated fuel samples.	3.28
	Isotopic composition and Pu content in mixed-oxide fuel depend strongly on the discharge burnup of spent fuel assemblies from which plutonium has been extracted. The ratio of fissile isotopes for the plutonium also varies; this will affect the characteristics of the reactor core. In addition, the Pu vector (Pu-238, Pu-239, Pu-240, Pu-241 and Am-241) needs to be incorporated in the mixed-oxide core design, recognizing that there are changes affecting reactivity and key neutronics parameters as a function of the start-up time after mixed-oxide fuel fabrication. These features need to be taken into account in core design and safety analyses.	3.128 (a)
Fuel channel	In boiling water reactors, the pressure difference between the inside and outside of the boundary of the fuel channel could induce bowing and bulging of the fuel channel. This deformation, as well as fuel cladding bowing, could consequently increase the local flux peaking and cause friction affecting control assembly movement.	3.43 (a)
Load following	Load following means that the reactor will be operated so that electricity generation will match a varying electrical demand. Load following implies operation with power manoeuvers at levels less than the rated thermal power, so the total amount of electrical energy output is less than if the unit is operated at a relatively	3.129

Topics	Clarification	Supplement to paragraph no.
	constant base load. Load following operation could require increased maintenance and monitoring and may complicate the reliability and aging assessments of some structures, systems and components	
	During load following operation, power density redistributions are caused promptly by control rod movement, but then enable inherent subsequent redistribution processes via feedback effects linked with the reactor coolant conditions and the xenon distributions. This generates power density distribution changes characterized by higher peak power densities (and/or lower departure from nucleate boiling ratios) compared with initial unperturbed conditions.	3.132
Margin	In the context of this Safety Guide, "margin" refers to the difference between the design limit defined for a specific physical parameter and the extremum (minimum or maximum) value of this physical parameter. ~~This difference is called "design margin".~~ The terminology "design limits" is absent in the IAEA Safety Glossary ([9]) but is used in SSR-2/1 (Rev 1) [1]. ~~In case the failure limit for this specific physical parameter is known, the difference between the design limit and the failure limit is called "safety margin".~~ "Design limits" are used in this Safety Guide ~~interchangeably~~ to deal with ~~the~~ commonly used "safety limits", "~~safety~~ operational limits" ~~or~~ and "acceptance criteria" defined in the IAEA Safety Glossary (~~Ref.~~ [9]).	2.12
	The term "shutdown margin" is not defined in the IAEA Safety Glossary [9]; however, it is generally accepted as the instantaneous amount of reactivity by which a reactor remains subcritical from its present conditions assuming all full-length ~~rod cluster assemblies~~control rod assemblies are fully inserted except for the ~~single rod cluster assembly~~one ~~of~~ exhibiting the highest reactivity worth that is assumed to be fully withdrawn.	3.18

Topics	Clarification	Supplement to paragraph no.
Pellet-cladding interaction	The cladding creepdown and fuel pellet thermal expansion and gaseous swelling will lead to strain-driven pellet-cladding mechanical interaction during all applicable plant states. Failure mode via pellet-cladding mechanical interaction is by cladding ductility exhaustion.	3.45
	Stress corrosion cracking in the fuel cladding occurs when the stresses on the inner surface of the cladding (as a result of pellet-cladding interaction) reach a certain limit under a corrosive environment. After a power reduction, the thermal contraction of the fuel pellets causes re-opening the pellet-cladding gap (or the gaps between the pellets fragments). If the reduced power operation is maintained long enough (i.e., extended reduced power operation), the fuel cladding will creep down and close the gaps again. The fuel ~~element~~ rod is then considered as re-conditioned at this lower power level. When the reactor core returns to full power at a later time, tensile stresses will appear in the cladding. These residual stresses will increase the susceptibility to stress corrosion cracking driven by pellet-cladding interaction under corrosive fission product environments in the fuel ~~element~~rod.	3.46
	The power-ramp failure threshold is the lowest within a burnup range called the “critical burnup range”. For fuel burnup values below this “critical burnup range”, the pellet-cladding gap remains open, so that the power change has to be larger to reach the same level of stress in the cladding as compared to a closed gap condition. For fuel burnup values above this “critical burnup range”, experience shows that the pellet-cladding interfacial material compound generated by irradiation is such that the stress concentration on the inner surface of the cladding is reduced, making stress corrosion cracking in the cladding unlikely. Since the “critical burnup range” depends on pellet-cladding gap closure kinetics, it is dependent upon the specific material properties of the	3.47

Topics	Clarification	Supplement to paragraph no.
	cladding type and fuel ~~element~~ rod design.	
Reactivity feedbacks	The inherent neutronic characteristics of the reactor core are typically characterized by the following reactivity feedbacks or coefficients: (a) Reactivity feedback due to fuel temperature changes (i.e., temperature coefficient of reactivity for the fuel orDoppler coefficient); (b) Reactivity feedback due to coolant/moderator temperature changes, including related coolant/moderator density (i.e., temperature coefficients of reactivity for the coolant and the moderator); (c) Reactivity feedback due to changes in the void fraction in the coolant/moderator (i.e., void coefficients of reactivity for the coolant and the moderator); (d) The delayed neutron fraction and the prompt neutron lifetime, (e) The effects of power redistribution on reactivity (e.g. the xenon efficiency and the moderator density); and (f) Decay of xenon and other neutron absorbers in the long term core analysis;	3.11
Shutdown	The simplest common form of design for fail-safe shutdown allows the shutdown devices to be held above the core by active means. Provided that the guide structures for the shutdown devices are not obstructed, the devices will drop into the core under gravity in the event of a de-energization of the active means of holding them (e.g. a loss of current through a holding electromagnet). This does not apply to boiling water reactors.	3.88 (c)
	The overall reactivity worth of the shutdown devices is a function of the spacing between the devices, as well as of their locations in	3.91 (f)

Topics	Clarification	Supplement to paragraph no.
	the reactor. When two devices are close together, their worth is less than the sum of their individual worths.	

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